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RBG-46052

December 18, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: River Bend Station, Unit 1
Docket No. 50-458
License Amendment Request
Revised Full-Scope Application of NUREG-1465 Alternative Source Term Insights

Reference: 1. Letter RBG-45930 from P. D. Hinnenkamp to U.S. NRC Document Control Desk, "License Amendment Request – Full Scope Application of NUREG-1465 Alternative Source Term Insights", dated April 24, 2002.

Dear Sir or Madam.

By letter (reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the River Bend Station (RBS) Technical Specifications (TSs) to apply alternative source term insights. This letter revises that license amendment request. This revision reduces the scope of the original request and applies the term "recently irradiated" in areas that were previously proposed for deletion. Typographical errors identified in the original submittal are also corrected. Changes are indicated by revision bars in the margins. The information provided in supplements to RBG-45930 remains valid and is not impacted by this revision. The No Significant Hazards Consideration conclusions are not changed by this revision. Additionally, Attachment 4 is changed to delete the commitment contained in the original submittal, as it is no longer applicable. New commitments to address control room habitability concerns are contained in supplemental letter RBG-46053. Enclosure 1 is also provided containing a CD Rom with electronic files requested by the staff.

On December 23, 1999, the NRC published new regulation 10 CFR 50.67, "Accident Source Term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the traditional source term used in design-basis accident analyses with an alternative source term. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequences analyses should apply for a license amendment under 10 CFR 50.90.

Regulatory guidance for the implementation of the alternative source term is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000. This regulatory guide provides guidance to licensees of operating nuclear plants on acceptable applications of alternative

A001

source terms. The use of an alternative source term changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents.

The alternative source term analyses for River Bend Station were performed following the guidance in accordance with Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The analyses included the control rod drop, fuel handling, loss of coolant, and main steam line break accident scenarios.

The attachments provided by reference 1 have been revised and are included in their entirety with revision bars that denote changes. Attachment 1 contains Entergy's Analysis of Proposed Technical Specification Change. Attachment 2 contains the marked-up pages of the Technical Specifications showing the proposed changes. Attachment 3 consists of a marked-up copy of the Technical Specification Bases associated with this proposed change. Attachment 4 provides a commitment identification form. Attachment 5 provides a Post-LOCA Suppression Pool pH Evaluation Summary. Attachment 6 provides the results of Atmospheric Dispersion Factors (χ/Q) Calculations. Attachment 7 provides a Loss of Coolant Accident (LOCA) Dose Analysis Summary. Attachment 8 provides a Fuel Handling Accident (FHA) and Light Load Drop Accident (LLA) Summary. Attachment 9 provides Control Rod Drop Accident (CRDA) Summary and Attachment 10 provides a Main Steam Line Break (MSLB) Outside of Primary Containment Summary. Enclosure 1 provides a CD Rom containing electronic files requested by the staff.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

The NRC has recently approved a number of TS Amendments which fully utilize the AST methodology. Entergy has reviewed the previously approved amendments and information submitted to incorporate any lessons learned.

Entergy requests approval of the proposed amendment by February 10, 2003. This request date will allow time for implementation prior to RF11, which will occur in the Spring of 2003. Once approved, the amendment shall be implemented within 60 days. Although this request is neither exigent nor emergency, your prompt review is requested.

If you have any questions or require additional information, please contact Greg Norris at 225-336-6391.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 18, 2002.

Sincerely,

Rick J. King
Director, Nuclear Safety Assurance
River Bend Station, Unit 1
RJK/GPN

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to TS Bases pages (Information Only)
4. List of Regulatory Commitments
5. Post-LOCA Suppression Pool pH Evaluation Summary
6. Atmospheric Dispersion Factors (χ/Q) Calculations
7. Loss of Coolant Accident (LOCA) Dose Analysis Summary
8. Fuel Handling Accident (FHA) and Light Load Drop Accident (LLA) Summary
9. Control Rod Drop Accident (CRDA) Summary
10. Main Steam Line Break (MSLB) Outside of Primary Containment Summary

Enclosure:

1. CD containing electronic files requested by the staff

cc: U. S. Nuclear Regulatory Commission
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File Nos.: G9.5, G9.42

File: LAR 2001-37

File: RBF1-02-0204

Attachment 1

Letter RBG-46052

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-47 for River Bend Station, Unit 1 (RBS). The proposed changes will revise the Operating License and design bases to reflect application of alternative source term methodology (AST).

This submittal represents a full-scope implementation of the new source term. Design basis accident analyses have been revised to define the impact of the new source term on doses to the public at the site boundary and to the operator in the control room. The analyses also bounds the Thermal Power Optimization (Appendix K power uprate) presently being planned for River Bend Station (RBS).

The accident source term is a significant aspect of the design and licensing basis of the plant. As an input to the accident analyses that form the basis for the design and operation of the unit, a change in the source term can impact both the postulated accident consequences and the margin of safety. For this reason, the NRC has determined that any change to the design basis to use an alternative source term should be reviewed and approved by the NRC in the form of a license amendment. No specific plant modifications are required to implement alternative source term. Any modification that utilizes the benefits of alternative source term will be evaluated under 10 CFR 50.59 as allowed per Regulatory Guide 1.183.

Entergy requests approval of the proposed amendment by February 10, 2002. This request date will allow time for implementation prior to Refuel Outage 11 (RF11), which is scheduled for the Spring of 2003.

2.0 PROPOSED CHANGES AND TECHNICAL ANALYSIS

Technical Specification Section: 1.1, "Definitions"

Description of Change:

- Re-define "Dose Equivalent I-131" and delete reference to TID-14844

Justification:

The current definition of Dose Equivalent I-131 refers to TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," which is the basis for the previous dose methodology used at RBS. This definition is directly applicable to the maximum allowable coolant concentrations Technical Specification (TS) Surveillance Requirement (SR) contained in TS 3.4.8, "RCS Coolant Activity." The Bases for that TS explains that the purpose of the surveillance is to ensure that the source term potentially released is bounded by that assumed in the Main Steam Line Break (MSLB) analysis. The MSLB analysis (Attachment 10) used dose conversion factors from Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, in scaling the core concentrations to the TS maximum values. Therefore, the reference to TID-14844 is no longer appropriate and must be deleted.

Technical Specification Section: 3.3.6.2, "Secondary Containment and Fuel Building Isolation Instrumentation"

Description of Change:

- No changes to TS 3.3.6.2 are requested.

All proposed changes to TS 3.3.6.2 requested by Reference 6.1 are withdrawn

Technical Specification Section: 3.3.7.1, "Control Room Fresh Air (CRFA) System Instrumentation"

Description of Change:

- Revise Note (b) in Table 3.3.7.1-1 to reflect that the requirements are only applicable during movement of "recently irradiated fuel."

Justification:

Filtration by the CRFA system is currently credited in all of the Design Basis Accidents: LOCA, FHA, Control Rod Drop Accident (CRDA), and MSLB. These analyses credit initiation of the system via either a LOCA signal (reactor water level 2, high drywell pressure, etc.) or via a high radiation signal from the intake radiation monitors. These signals tied with automatic initiation of the CRFA filters are the basis for this TS. The Bases for this section states "The ability of the CRFA System to maintain the habitability of the MCR is explicitly assumed for certain accidents as discussed in the USAR safety analyses."

The FHA (including the Light Load Drop Accident (LLA) analysis) and MSLB dose analyses do not credit filtration by the CRFA charcoal filters. However, those analyses assume a minimum decay time of 24 hours prior to an FHA potentially occurring. Currently "recently irradiated fuel" is defined in the TS bases (Section 3.6.4.5) as "fuel which has been part of a critical reactor core within the previous 11 days." The FHA analysis prepared in support of AST effectively redefines "recently irradiated fuel" based on a 24 hour decay time. Therefore, in the unlikely event fuel may be moved prior to 24 hours this specification would directly apply and the MCR Local Air Intake radiation monitors would be required to ensure that the potential dose to MCR operators meets 10CFR50.67 dose limits. It should be noted that a decay time of 24 hours is currently required prior to fuel movement per Technical Requirements Manual (TRM) Section 3.9.10.

One CRDA scenario credited the CRFA system, however, it assumed manual initiation of the system 20 minutes into the event. Both air intakes for the main control room have redundant radiation monitors which annunciate in the main control room. Automatic initiation of the system is not assumed. The calculated dose consequences of a CRDA meet the acceptance criteria from 10CFR50.67 and Regulatory Guide 1.183. The CRFA filters are also credited in the LOCA dose analysis (Attachment 7). That analysis also conservatively assumes manual initiation of the system, however, no relaxation of the automatic initiation requirements of the system is

requested at this time. All doses meet the acceptance criteria set forth in 10CFR50.67 and Regulatory Guide 1.183.

Technical Specification Section: 3.6.1.2, "Primary Containment Air Locks"

Description of Change:

- Revise SR 3.6.1.2.1 to remove requirements for annulus bypass leakage.
- Revise SR 3.6.1.2.4 to increase the leakage rate for the air lock seal pneumatic system from 1.28 psig/day to 1.5 psig/day.

Justification:

Annulus bypass leakage is leakage which will bypass the annulus and be released into the Auxiliary Building. All leakage paths (including the containment/fuel building personnel air lock and Inclined Fuel Transfer System [IFTS] drain line) which can potentially bypass secondary containment are currently included in the Secondary Containment Bypass (SCB) summation (See TRM Table 3.6.1.1-1 for a list of Annulus Bypass penetrations). Since the "annulus bypass" leakage paths now lead to the Auxiliary Building, the air would be filtered by SGTS prior to release to the environment. The AST LOCA dose analysis does not assume an explicit "annulus bypass" leakage path. RBS performed a sensitivity study demonstrating that off-site doses are not sensitive to annulus bypass leakage as long as such leakage paths are filtered. Annulus bypass leakage is only a small potion of the overall containment leakage term and both the Auxiliary Building and the Annulus are filtered by SGTS. Therefore, the impact to calculated AST doses is negligible and simplifying the model is acceptable. Thus, the annulus bypass leakage summation is no longer required and should be deleted from Technical Specifications.

The primary containment personnel air locks (PAL) are used to ingress and egress the primary containment. The PAL doors have a seal which remains pressurized via the seal air system. Technical Specification SR 3.6.1.2.2 states that the minimum normal operation pressure for the primary containment air lock seal air flask pressure must be ≥90 psig. The inflatable seals must remain above a pressure of 45 psig to maintain its integrity. The current allowable leakage rate of 1.28 psi per 24 hours is based on a period of 35 days ((90psig-45psig)/35days=1.286psi/day). Since dose calculations are performed for a period of 30 days a value of 1.5 psi/day is requested ((90-45)/30=1.5).

Technical Specification Section: 3.6.1.3, "Primary Containment Isolation Valves"

Description of Change:

- Revise SR 3.6.1.3.9. Specifically, increase the allowable secondary containment bypass leakage rate from "≤170,000 cc/hr when pressurized to $\geq P_a$" to "≤580,000 cc/hr when pressurized to $\geq P_a$."
- Revise SR 3.6.1.3.10 to set a single main steam line leakage limit of 50 scfh when tested at $\geq P_a$
- Delete SR 3.6.1.3.12 (annulus bypass leakage rate summation).

Justification:

Secondary containment bypass leakage is leakage from the primary containment building which will bypass the annulus and auxiliary building, thus, it will potentially escape to the environment unfiltered. This leakage is independent of the overall containment rate summation (L_a). Prior to RBS TS Amendment 98 the Penetration Valve Leakage Control System (PVLCS) system was assumed to be manually initiated which terminated this release path early in the event. The PVLCS system was deleted via TS Amendment 98. The Inclined Fuel Transfer System (IFTS) drain line and the fuel building personnel air lock were added since the fuel building was removed from the secondary containment envelope via RBS TS Amendment 113. A review of the potential leakage paths for this summation concluded that a value of 580,000 cc/hr was appropriate for use in the AST LOCA Analysis (Attachment 7). The analysis dose consequences using this value met the criteria set forth in 10CFR50.67.

Annulus bypass leakage is leakage which will bypass the annulus and be released into the Auxiliary Building. All leakage paths (including the containment/fuel building personnel air lock and IFTS drain line) which can potentially bypass secondary containment are currently included in the Secondary Containment Bypass (SCB) summation (TS SR 3.6.1.3.9). Since the "annulus bypass" leakage paths now lead to the Auxiliary Building, the air would be filtered by SGTS prior to release to the environment. The AST LOCA dose analysis does not assume an explicit "annulus bypass" leakage path. RBS performed a sensitivity study demonstrating that off-site doses are not sensitive to annulus bypass leakage as long as such leakage paths are filtered. Annulus bypass leakage is only a small portion of the overall containment leakage term, and both the Auxiliary Building and the Annulus are filtered by SGTS. Therefore, the impact to calculated AST doses is negligible and simplifying the model is acceptable. Thus, the annulus bypass leakage summation is no longer required and should be deleted from Technical Specifications.

The current TID dose analysis does not assume a failure of an MSIV. The AST analysis conservatively assumed that one MSIV failed (in addition to an EDG). The leakage rate of 50 scfh was assumed through the steam line containing the failed MSIV. This value corresponds to the proposed limit to be incorporated in SR 3.6.1.3.10.

Technical Specification Section: 3.6.1.10, "Primary Containment – Shutdown"

Description of Change:

- No changes to TS 3.6.1.10 are requested.

  All proposed changes to TS 3.6.1.10 requested by Reference 6.1 are withdrawn.

Technical Specification Section: 3.6.4.1, "Secondary Containment - Operating"

Description of Change:

- Revise SR 3.6.4.1.4. Specifically, revise the Auxiliary Building drawdown time from 13.5 seconds to 34.5 seconds.

Justification:

LOCA dose analyses do not credit secondary containment until an adequate vacuum is reached (≤-0.25 in. w.g.). The AST LOCA dose analysis (Attachment 7) conservatively assumed manual initiation of SGTS, even though the automatic initiation function and associated TS requirements are retained. The analysis also allowed an additional 10 minutes for an adequate vacuum to be established. This resulted in a total assumed Positive Pressure Period (PPP) of 30 minutes (20 minutes for Operator action and an additional 10 minutes to establish the required vacuum). Calculations used the GOTHIC computer code to demonstrate that the PPP assumed in the AST LOCA dose analysis are conservative for the annulus and auxiliary building, respectively. They determined a revised drawdown time for testing during non-accident conditions using current system parameters (flow rates, etc.). That analysis determined that an auxiliary building drawdown time of 38.5 seconds will ensure that the calculated PPP is conservative. This submittal requests a value of 34.5 seconds (~90% of the calculated value) for conservatism.

Technical Specification Section: 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs) and Fuel Building Isolation Dampers (FBIDs)"

Description of Change:

- No changes to TS 3.6.4.2 are requested.

  All proposed changes to TS 3.6.4.2 requested by Reference 6.1 are withdrawn.

Technical Specification Section: 3.6.4.5, "Fuel Building"

Description of Change:

- No changes to TS 3.6.4.2 are requested.

  All proposed changes to TS 3.6.4.5 requested by Reference 6.1 are withdrawn.

Technical Specification Section: 3.6.4.7, "Fuel Building Ventilation System – Fuel Handling"

Description of Change:

- No changes to TS 3.6.4.7 are requested.

  All proposed changes to TS 3.6.4.7 requested by Reference 6.1 are withdrawn.

Technical Specification Sections: 3.6.5.1, "Drywell" and 3.6.5.2, "Drywell Air Locks"

Description of Change:

- Revise SR 3.6.5.1.2 to increase the leakage rate for the air lock seal pneumatic system from 0.67 psig/day to 20.0 psig/day.

- Revise SR 3.6.5.2.5 to increase the leakage rate for the air lock seal pneumatic system from 0.67 psig/day to 20.0 psig/day.

Justification:

Drywell integrity is credited in the containment and drywell pressure response analyses developed in support of the power uprate project approved via TS Amendment 114. The current leakage rate is based on a 30 day duration, i.e., drywell seal integrity is required for at least 30 days following a postulated LOCA. The proposed leakage rate for the drywell air lock seals is based on leakage over the first 24 hours. After 24 hours, the seals could potentially fail due to the internal pressure not being adequate. A large break in the drywell would increase pressure rapidly and uncover the suppression pool vents. Containment pressure would increase steadily as a result of flow from the drywell. Following this initial blowdown period, the pressure in the drywell begins to drop due to steam condensation. Eventually the drywell pressure drops below the containment pressure and becomes a relative vacuum (see figures in RBS USAR Section 6.2). This occurs well before 24 hours. Failure of the seals would tend to reduce containment pressure to less than what is analyzed since the drywell is at a lower pressure. Thus, the analyses for large breaks are conservative since the pressures they calculate bound the pressures that would be expected if the drywell air lock seals failed.

Intermediate and Small Break Accidents were also evaluated. The "short term" analyses show the drywell pressure is greater than containment pressure, however, the "long term" (>10,000 seconds) analyses show the drywell pressure drops below containment pressure well before 24 hours. The reason for the differences lies with computer code limitations and decay heat assumptions. Since the drywell has a lower pressure than the containment 24 hours into the event, the calculated containment pressures are conservative and bound seal failure at 24 hours.

The AST LOCA dose analysis (Reference Attachment 7) treats the drywell and containment as two separate nodes. Early in the event (<20 minutes) the flow rate between the drywell and containment is based on the containment pressure response to a recirculation line break. Regulatory Guide 1.183, Appendix A, Section 3.7 states "After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment." Since the containment and drywell are homogenized at 2 hours as required by Regulatory Guide 1.183, and seal failure would not occur until after 24 hours, there is no impact to the LOCA doses calculated using AST based assumptions.

The air lock seals remain pressurized via the seal air system. Technical Specifications SR 3.6.5.1.1 and SR 3.6.5.2.2 require that the minimum normal operating pressure for drywell seal air flask pressure must be ≥75 psig. The inflatable seals must remain above a pressure of 55 psig to maintain its integrity. The current allowable leakage rate of 0.67 psi per 24 hours is based on a period of 30 days ((75psig - 55psig)/30days = 0.67 psi/day). The discussion above demonstrates that the seals are not required after 24 hours, therefore, a value of 20.0 psi per 24 hours is requested ((75psi - 55psi)/1day= 20.0 psi/day).

Technical Specification Section: 3.7.2, "Control Room Fresh Air (CRFA) System"

Description of Change:

- Revise the APPLICABILITY from "during movement of irradiated fuel..." to "during movement of recently irradiated fuel..."
- Revise CONDITION C and associated REQUIRED ACTIONS to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."
- Revise CONDITION E and associated REQUIRED ACTIONS to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

Justification:

Filtration by the CRFA system is currently credited in all of the Design Basis Accidents: LOCA, FHA, CRDA, and MSLB. These analyses credit initiation of the system via either a LOCA signal (reactor water level 2, high drywell pressure, etc.) or via a high radiation signal from the intake radiation monitors. These signals tied with automatic initiation of the CRFA filters is the basis for this TS.

The MSLB dose analyses do not credit filtration by the CRFA charcoal filters. The FHA analysis also does not credit the CRFA charcoal filters. However, it inherently assumes a decay time of 24 hours prior to an FHA potentially occurring. Therefore, during movement of "recently irradiated fuel" the TS and applicable SR will be applicable since operation of the charcoal filter train would be required to ensure applicable doses are bounded. As such the definition of "recently irradiated fuel" will be tied to the 24 hour decay time assumed in the AST FHA analysis. It should be noted that movement of irradiated fuel with decay times <24 hours is currently prohibited at RBS.

The CRFA filters are credited in the LOCA dose analysis (Reference Attachment 7). That analysis assumes manual initiation of the system. Review of system design indicates that in actuality the CRFA system will be operating in emergency mode prior to the onset of fuel damage. Specifically, the current TID LOCA analysis assumes that the system starts automatically and credits ESF charcoal filtration 66 seconds into the event, whereas the AST dose methodology does not assume fuel damage until 2 minutes into the event. Therefore, assuming manual initiation of the system at 20 minutes is clearly conservative.

The CRDA analysis (Reference Attachment 9) also credited the CRFA system, however, it conservatively assumed manual initiation rather than automatic initiation of the charcoal filters. Fuel damage for a CRDA is assumed to occur at the onset of the event. Once again, however, the 20 minutes assumed for manual initiation of the system easily bounds the initiation time assumed in the current CRDA analysis of 66 seconds.

Technical Specification Section: 3.7.3, "Control Room AC System"

Description of Change:

- Revise the APPLICABILITY statement from "during movement of irradiated fuel..." to "during movement of recently irradiated fuel..."
- Revise the CONDITION D and associated REQUIRED ACTIONS to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."
- Revise the CONDITION E and associated REQUIRED ACTIONS to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

Justification:

Technical Specification Bases for Section 3.7 explains that the intent of the section is to ensure that required equipment is not affected by adverse environmental conditions, specifically, high temperatures. The applicability discussion states that "In Modes 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these modes. Therefore, maintaining the Control Room AC System OPERABLE is not required in Modes 4 and 5 except when a significant radiological release is available. The FHA no longer credits mitigation by any system including the CR AC and CRFA Systems. However, a decay time of 24 hours is assumed, therefore, the TS would remain applicable during movement of "recently irradiated fuel." The dose consequences meet the criteria set forth in 10CFR50.67 and Regulatory Guide 1.183. Therefore, this TS is no longer applicable.

Technical Specification Section: 3.8.2, "AC Sources – Shutdown"

Description of Change:

- Revise the APPLICABILITY statement from "during movement of irradiated fuel..." to "during movement of recently irradiated fuel..."
- Revise the REQUIRED ACTIONS associated with CONDITION A to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."
- Revise the REQUIRED ACTIONS associated with CONDITION B to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

Justification:

The TS Bases for this section explains that this TS is applicable to ensure that "Systems needed to mitigate a fuel handling accident are available." Previously the CRFA and fuel building ventilation systems' charcoal filter trains were credited to mitigate the consequences of a FHA. The AST analyses do not credit these or any other systems to mitigate the consequences of a FHA. Based on the above discussion it can be concluded that accident assumptions will be met, therefore, this TS is no longer applicable except during movement of "recently irradiated fuel."

Technical Specification Section: 3.8.5, "DC Sources – Shutdown"

Description of Change:

- Revise the APPLICABILITY statement from "during movement of irradiated fuel..." to "during movement of recently irradiated fuel..."
- Revise the REQUIRED ACTIONS associated with CONDITION A to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

Justification:

The TS Bases for this section explains that this TS is applicable to ensure that "Systems needed to mitigate a fuel handling accident are available." Previously the CRFA and fuel building ventilation systems' charcoal filter trains were credited to mitigate the consequences of a FHA. The AST analyses do not credit these or any other systems to mitigate the consequences of a FHA. Based on the above discussion it can be concluded that accident assumptions will be met, therefore, this TS is no longer applicable except during movement of "recently irradiated fuel."

Technical Specification Section: 3.8.8, "Inverters – Shutdown"

Description of Change:

- Revise the APPLICABILITY statement from "during movement of irradiated fuel..." to "during movement of recently irradiated fuel..."
- Revise the REQUIRED ACTIONS associated with CONDITION A to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

Justification:

The TS Bases for this section explains that this TS is applicable to ensure that "Systems needed to mitigate a fuel handling accident are available." Previously the CFRA and fuel building ventilation systems' charcoal filter trains were credited to mitigate the consequences of a FHA. The AST analyses do not credit these or any other systems to mitigate the consequences of a FHA. Based on the above discussion it can be concluded that accident assumptions will be met, therefore, this TS is no longer applicable except during movement of "recently irradiated fuel."

Technical Specification Section: 3.8.10, "Distribution Systems – Shutdown"

Description of Change:

- Revise the APPLICABILITY statement from "during movement of irradiated fuel..." to "during movement of recently irradiated fuel..."
- Revise the REQUIRED ACTIONS associated with CONDITION A to be consistent with the APPLICABILITY change concerning movement of "recently irradiated fuel."

Justification:

The TS Bases for this section explains that this TS is applicable to ensure that "Systems needed to mitigate a fuel handling accident are available." Previously the CFRA and fuel building ventilation systems' charcoal filter trains were credited to mitigate the consequences of a FHA. The AST analyses do not credit these or any other systems to mitigate the consequences of a FHA. Based on the above discussion it can be concluded that accident assumptions will be met, therefore, this TS is no longer applicable except during movement of "recently irradiated fuel."

Technical Specification Section: 3.10, "Special Operations"

Description of Change:

- No changes to TS 3.10 are requested.

  All proposed changes to TS 3.10 requested by Reference 6.1 are withdrawn.

Technical Specification Section: 5.5.7, "Ventilation Filter Testing Program (VFTP)"

Description of Change:

- Revise the SGTS allowable penetration from 0.5% to 5.0%.
- Revise the CRFA allowable penetration from 0.5% to 1.0%.

Justification:

RBS committed to ASTM D3803-1989 via Technical Specification Amendment 115 which allows licensees to test to 50% of the margin assumed in the safety analyses. SGTS is only credited in the LOCA doses analysis which assumes 90% filter efficiency, therefore, the testing acceptance criteria is being revised to 5%. The CRFA filters are credited in the LOCA and the CRDA analyses. These analyses assume a filter efficiency of 98% which mandates a testing acceptance criteria of 1%. The changes proposed with respect to the Fuel Building ventilation system in reference 1 are withdrawn.

Technical Specification Section: 5.5.13, "Primary Containment Leakage Rate Testing Program"

Description of Change:

- Revise 5.5.13 to increase the containment leakage rate from 0.26% per day to 0.325% per day.

Justification:

The containment leakage rate is a major assumption in the LOCA dose analysis. The AST LOCA analysis (Reference Attachment 7) assumed a containment leakage rate of 0.325% per day. The dose consequences of that analysis met the criteria set forth in 10CFR50.67, therefore, the increased leakage rate is acceptable.

3.0 BACKGROUND

The current dose methodology used at River Bend Station is based on NRC guidance from the 1960's and 1970's (Regulatory Guide 1.3, TID-14844, etc.). Due to the simplistic modeling techniques and limited knowledge at that time, many assumptions are grossly conservative which led to the stringent testing requirements presently contained in the RBS Technical Specifications (TS).

The Alternate Source Term (AST) project is the result of rigorous analyses performed to more accurately model the evolution of Severe Accidents. Since the methodology is more realistic, the dose consequences utilizing this methodology are often lower than the Design Basis

Accidents scenarios' overly conservative assumptions. The NRC has recently approved several TS Amendments which fully utilize the AST methodology.

Regulatory Guide 1.183 was developed to provide regulatory guidance in utilizing the AST dose methodology. Analyses were developed to be consistent with the requirements provided within that Regulatory Guide. Initial application of AST to a plant must be reviewed and approved by the NRC. Once initial AST implementation has been approved by the staff and is part of the facility design basis, licensees may use 10CFR50.59 in assessing safety margins related to plant modifications and changes to procedures. Reg. Guide 1.183 allows for partial or full implementation of AST.

Note that a limited scope application of AST was previously approved by the NRC for RBS. Specifically, AST was used to justify neglecting the Inclined Fuel Transfer System (IFTS) drain line as a source to operators stationed to manually close an isolation valve. See RBS Technical Specification Amendment 116 for more information.

4.0 TECHNICAL ANALYSIS

All of the major accidents (which could potentially lead to significant off-site doses) contained in USAR Chapter 15 were reanalyzed using the AST dose methodology. These events include the Control Rod Drop Accident (CRDA as contained in USAR Section 15.4.9), the Main Steam Line Break Outside of Containment (MSLB, USAR Section 15.6.4), the Loss of Coolant Accident (LOCA, USAR Section 15.6.5), and the Fuel Handling Accident (FHA, USAR Chapter 15.7.4). A number of supporting analyses, such as the Light Loads Analysis, were also reanalyzed using AST assumptions. Regulatory Guide 1.183 provides guidance in applying the AST insights found in NUREG-1465 to plant analyses. Attachments 7 through 10 of this submittal provide the summaries of these analyses.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. The analyses used the assumptions and guidance provided by Regulatory Guide 1.183. No exceptions to Regulatory Guide 1.183 assumptions were taken for the FHA, CRDA, or MSLB analyses. The LOCA analysis had one deviation from Regulatory Guide 1.183, Appendix A assumptions. This deviation concerns the assumed liquid leakage rate from Engineered Safety Features (ESF) Emergency Core Cooling Systems (ECCS). This deviation is discussed in detail in Attachment 7 and is similar to assumptions contained in the current (TID) LOCA dose analysis.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the SAR.

5.2 No Significant Hazards Consideration

This proposed amendment to the River Bend Technical Specifications (TS) revises those specifications affected by the implementation of the alternative source term concepts in accordance with NUREG 1465. In addition, based on the alternative source term, changes are proposed to selected specifications associated with handling irradiated fuel in the primary containment or Fuel Building and CORE ALTERATIONS. The alternative source term changes affect the definitions, and the specifications for the Control Room Fresh Air System, Standby Gas Treatment System, and leakage rates for Primary Containment and the Personnel Airlocks seal air systems.

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The alternative source term does not require modification of the facility; rather, once the occurrence of an accident has been postulated the new source term is an input to evaluate the potential consequences. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at River Bend Station. Based on the results of these analyses, it has been demonstrated that, even with the requested Technical Specification changes, the dose consequences of these limiting events are within the regulatory guidance currently approved by the NRC for use with the alternative source term. This guidance is presented in Regulatory Guide 1.183, 10CFR50.67 and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

   Because the equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident. The proposed requirements bound the conditions of the current design basis fuel handling accident analysis which concludes that the radiological consequences are within the acceptance criteria of NUREG 0800, Section 15.7.4 and General Design Criteria 19. As noted above, with the alternative source term implementation, the acceptance criteria are also being revised. The results of the revised Fuel Handling Accident demonstrate that the dose consequences are within the NRC regulatory guidance. This guidance is presented in Regulatory Guide 1.183, 10CFR50.67 and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

The proposed changes using the alternative source term dose methodology are analytical in nature and do not physically alter the facility or of any equipment within the facility. Similarly, the alternative source term does not create any new initiators or precursors of a new or different kind of accident. The proposed changes to the Technical Specifications, while they revise certain performance requirements, do not involve any physical modifications to the plant.

The proposed changes related to shutdown controls based on the alternative source term do not create the possibility of a new or different kind of accident from any previous analyzed.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The changes above are associated with the implementation of a new licensing basis for River Bend Station. Approval of the basis change from the original source term in accordance with TID-14844 to the new alternative source term of NUREG-1465 is requested by this submittal. The results of the accident analyses prepared in support of this submittal are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies as outlined in the regulatory guidance and conservatively represent the requested Technical Specification changes. Safety margins and analytical conservatisms have been evaluated and are well understood. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound all postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in Regulatory Guide 1.183, 10CFR50.67 and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries as well as control room, are within the corresponding regulatory limits. In a similar way, the results of the existing analyses demonstrated that the dose consequences were within the applicable NRC-specified regulatory limit. Specifically, the margin of safety for these accidents is considered to be that provided by meeting the applicable regulatory limit for Alternate Source Term methodologies, which, for most events, is conservatively set at, or below, the 10CFR50.67 limit. With respect to the control room personnel doses, the margin of safety is the difference between the 10CFR100 limits and the regulatory limit defined by 10CFR50, Appendix A, General Design Criterion (GDC) 19.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 Letter RBG-45930 from P. D. Hinnenkamp to U.S. NRC Document Control Desk, "License Amendment Request – Full Scope Application of NUREG-1465 Alternative Source Term Insights", dated April 24, 2002.

6.2 Letter RBG-45989 from Rick J. King to U.S. NRC Document Control Desk, "Supplement to License Amendment Request - Full Scope Application of NUREG-1465 Alternative Source Term Insights", dated July 18, 2002

Attachment 2

Letter RBG-46052

Proposed Technical Specification Changes (mark-up)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."~~

(continued)

INSERT A

INSERT A

Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

Table 3 3 7.1-1 (page 1 of 1)
Control Room Fresh Air System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A 1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1	Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2	B	SR 3 3 7.1 1 SR 3 3 7.1 2 SR 3 3 7.1 3 SR 3 3 7.1 4 SR 3 3 7.1 5	≥ -47 inches
2	Drywell Pressure - High	1,2,3	2	C	SR 3 3 7.1.1 SR 3 3 7.1.2 SR 3 3 7.1.3 SR 3 3 7.1.4 SR 3 3 7.1.5	≤ 1.88 psid
3	Control Room Local Intake Ventilation Radiation Monitors	1,2,3 (a),(b)	1	D	SR 3 3 7 1.1 SR 3 3 7 1.2 SR 3 3 7 1.4 SR 3 3 7.1.5	≤ 0.97×10^{-5} µCi/cc

(a) During operations with a potential for draining the reactor vessel

(b) During the movement of recently radiated fuel assemblies in the primary containment or fuel building

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. ----------	
	Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program. ~~Verify the combined leakage rate is ≤ 13,500 cc/hr for all required annulus bypass leakage paths when pressurized to ≥ P_a.~~	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.2.2	Verify primary containment air lock seal air flask pressure is ≥ 90 psig.	7 days
SR 3.6.1.2.3	----------NOTE---------- Only required to be performed upon entry or exit through the primary containment air lock. ----------	
	Verify only one door in the primary containment air lock can be opened at a time.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.2.4	Verify, from an initial pressure of 90 psig, the primary containment air lock seal pneumatic system pressure does not decay at a rate equivalent to > ~~1.28~~ 1.50 psig for a period of 24 hours.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	Verify in-leakage rate of ≤ 340 scfh for each of the following valve groups when tested at 11.5 psid for MS-PLCS valves. a. Division I MS-PLCS valves b. Division II MS-PLCS valves	18 months
SR 3.6.1.3.9	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ ~~170,000~~ 580,000 cc/hr when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.10	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- ~~Verify~~ leakage rate through the valves served by each division of MS-PLCS is ≤ 150 scfh per division when tested at ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

Verify leakage rate through each main steam line is ≤ 50 scfh when tested at ≥ P_a, and the total

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.1.3.12 ~~Verify the combined leakage rate is ≤ 13,500 cc/hr for all required annulus bypass leakage paths when pressurized to ≥ P_a.~~ Deleted	~~In accordance with the Primary Containment Leakage Rate Testing Program~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed and loop seals filled.	31 days
SR 3.6.4.1.3	Verify each secondary containment access door is closed, except when the access opening is being used for entry and exit.	31 days
SR 3.6.4.1.4	Verify each standby gas treatment (SGT) subsystem will draw down the shield building annulus and auxiliary building to ≥ 0.5 and ≥ 0.25 inch of vacuum water gauge in ≤ 18.5 and ≤ ~~13.5~~ 34.5 seconds, respectively.	18 months on a STAGGERED TEST BASIS
SR 3.6.4.1.5	Deleted	Not Applicable
SR 3.6.4.1.6	Verify each SGT subsystem can maintain ≥ 0.5 and ≥ 0.25 inch of vacuum water gauge in the shield building annulus and auxiliary building, respectively, for 1 hour.	18 months on a STAGGERED TEST BASIS
SR 3.6.4.1.7	Deleted	Not Applicable

3.6 CONTAINMENT SYSTEMS

3.6.5.1 Drywell

LCO 3.6.5.1 The drywell shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell inoperable.	A.1 Restore drywell to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.1	Verify personnel door inflatable seal air flask pressure ≥ 75 psig.	7 days
SR 3.6.5.1.2	Verify from an initial pressure of 75 psig, the personnel door inflatable seal pneumatic system pressure does not decay at a rate equivalent to ≥ ~~0.67~~ 20.0 psig for a period of 24 hours.	18 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.2.1	Deleted	
SR 3.6.5.2.2	Verify drywell air lock seal air flask pressure is ≥ 75 psig.	7 days
SR 3.6.5.2.3	----------NOTE---------- Only required to be performed upon entry into drywell. ---------- Verify only one door in the drywell air lock can be opened at a time.	24 months
SR 3.6.5.2.4	Deleted	
SR 3.6.5.2.5	Verify, from an initial pressure of 75 psig, the drywell air lock seal pneumatic system pressure does not decay at a rate equivalent to > ~~0.67~~ 20.0 psig for a period of 24 hours.	18 months

3.7 PLANT SYSTEM

3.7.2 Control Room Fresh Air (CRFA) System

LCO 3.7.2 Two CRFA subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

During movement of recently irradiated fuel assemblies in the primary containment, or fuel building.

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRFA subsystem inoperable.	A.1 Restore CRFA subsystem to OPERABLE status.	7 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the primary containment or fuel building during or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	C.1 Place OPERABLE CRFA subsystem in emergency mode.	Immediately
	OR	
	C.2.1 Suspend movement of recently irradiated fuel assemblies in the primary and fuel building.	Immediately
	AND	
	C.2.2 Initiate action to suspend OPDRVs.	Immediately
D. Two CRFA subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CRFA subsystems inoperable during movement of recently irradiated fuel assemblies in the primary containment or fuel building, or during OPDRVs.	E.1 Suspend movement of recently irradiated fuel assemblies in the primary containment and fuel building.	Immediately
	AND	
	E.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Operate each CRFA subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.2.2	Perform required CRFA filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.2.3	Verify each CRFA subsystem actuates on an actual or simulated initiation signal.	18 months

(continued)

3.7 PLANT SYSTEMS

3.7.3 Control Room Air Conditioning (AC) System

LCO 3.7.3 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1 Restore control room AC subsystem to OPERABLE status.	30 days
B. Two control room AC subsystems inoperable.	B.1 Verify control room area temperature ≤ 104/F. AND B.2 Restore one control room AC subsystem to OPERABLE status.	Once per 4 hours 7 days
C. Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the primary containment or fuel building, or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	D.1 Place OPERABLE control room AC subsystem in operation.	Immediately
	OR	
	D.2.1 Suspend movement of recently irradiated fuel assemblies in the primary containment and fuel building.	Immediately
	AND	
	D.2.2 Initiate action to suspend OPDRVs.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies in the primary containment or fuel building, or during OPDRVs. recently	E.1 Suspend movement of irradiated fuel assemblies in the primary containment and fuel building. recently AND E.2 Initiate action to suspend OPDRVs.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources–Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems–Shutdown";

b. One diesel generator (DG) capable of supplying one division of the Division I or II onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10; and

c. One qualified circuit, other than the circuit in LCO 3.8.2.a, between the offsite transmission and the Division III onsite Class 1E electrical power distribution subsystem, or the Division III DG capable of supplying the Division III onsite Class 1E AC electrical power distribution subsystem, when the Division III onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.10.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

----------NOTE----------

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO Item a not met.	----------NOTE---------- Enter applicable Condition and Required Actions of LCO 3.8.10, when any required division is de-energized as a result of Condition A. ----------	
	A.1 Declare affected required feature(s) with no offsite power available from a required circuit inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the primary containment and fuel building.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. LCO Item b not met.	B.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	B.2 Suspend movement of irradiated fuel assemblies in primary containment and fuel building.	Immediately
	AND	
	B.3 Initiate action to suspend OPDRVs.	Immediately
	AND	
	B.4 Initiate action to restore required DG to OPERABLE status.	Immediately
C. LCO Item c not met.	C.1 Declare High Pressure Core Spray System and Standby Service Water System pump 2C inoperable.	72 hours

recently

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources–Shutdown

LCO 3.8.5 The following shall be OPERABLE:

a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division I or II onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown";

b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division I or II onsite Class 1E DC electrical power distribution subsystem(s) when required by LCO 3.8.10; and

c. The Division III DC electrical power subsystem capable of supplying the Division III onsite Class 1E DC electrical power distribution subsystem, when the Division III onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.10.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

----------NOTE----------

LCO 3.0.3 is not applicable

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies in the primary containment and fuel building.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

recently

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters–Shutdown

LCO 3.8.8 One Divisional inverter shall be OPERABLE capable of supplying one division of the Division I or II onsite Class 1E uninterruptible AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown".

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend handling of recently irradiated fuel assemblies in the primary containment or fuel building.	Immediately
	AND	(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems–Shutdown

LCO 3.8.10 The necessary portions of the Division I, Division II, and Division III AC, DC, and Division I and II AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the primary containment and fuel building.	Immediately
	AND	
		(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1989 at the system flowrate specified below ± 10%:

ESF Ventilation System	Flowrate
SGTS	12,500 cfm
FBVS	10,000 cfm
CRFAS	4,000 cfm

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below:

ESF Ventilation System	Penetration	RH
SGTS	5.0 % ~~0.5%~~	70%
FBVS	0.5%	70%
CRFAS	1.0 % ~~0.5%~~	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1989 at the system flowrate specified below ± 10%:

ESF Ventilation System	Delta P	Flowrate
SGTS	< 8" WG	12,500 cfm
FBVS	< 8" WG	10,000 cfm
CRFAS	< 8" WG	4,000 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that do not meet the criteria of either Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 DELETED

5.5.13 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 7.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.325% ~~0.26%~~ of primary containment air weight per day.

The Primary Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

The provisions of SR 3.0.2 do not apply to test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Attachment 3

Letter RBG-46052

Changes to Technical Specification Bases Pages
Information Only

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Primary Containment and Drywell Isolation

2.a. Reactor Vessel Water Level-Low Low, Level 2

50.67

Low RPV water level indicates the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ are not exceeded. The Reactor Vessel Water Level-Low Low, Level 2 Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA. In addition, Function 2.a provides an isolation signal to certain drywell isolation valves. The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the primary containment.

Reactor Vessel Water Level-Low Low, Level 2 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Low, Level 2 Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level-Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level-Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 1, 7, 8, 9, 15, and 16 valves. The isolation of valve Group 9 also includes the actuation of the Standby Gas Treatment System, the Control Room Fresh Air System, and the containment hydrogen analyzers.

2.b. Drywell Pressure-High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the automatic isolation valves on high

(continued)

B 3.3 INSTRUMENTION

B 3.3.6.2 Secondary Containment and Fuel Building Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation dampers (SCIDs) and starts appropriate ventilation subsystems. Similarly, the fuel building isolation instrumentation automatically initiates closure of appropriate fuel building isolation dampers (FBIDs) and initiates fuel building ventilation flow through the filtration system. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1), such that offsite radiation exposures are maintained within the requirements of 10 CFR ~~100~~ 50.67 that are part of the NRC staff approved licensing basis. Secondary containment isolation and establishment of vacuum within the assumed time limits ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment or during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits. Fuel building isolation ensures that fission products released due to fuel uncovery or a dropped fuel assembly are also maintained within regulatory limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) drywell pressure, and (c) fuel building ventilation exhaust radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation parameters. In addition, manual initiation of the logic is provided.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR ~~100~~ 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR ~~100~~ 50.67 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the USAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR ~~100.~~ 50.67

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR ~~100~~ 50.67 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of a limiting event while exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR ~~100~~ 50.67 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR ~~100.11~~. 50.67
2. USAR, Section 15.6.4.

BASES

BACKGROUND (continued)

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of ~~0.26%~~ 0.325% by weight of the containment and drywell air per 24 hours at the calculated maximum peak containment pressure (P_a) of 7.6 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

During plant operations in other than MODES 1, 2, and 3, the primary containment contains the fission products from a fuel handling accident (FHA), involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~11~~ 24 hours) inside the primary containment (Ref. 4), to limit doses at the site boundary to within limits. The primary containment air lock OPERABILITY assures a leak tight fission product barrier during activities with the unit shutdown.

Primary containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA, an FHA involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~11~~ 24 hours) or other unanticipated reactivity or water level excursion. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such events.

(continued)

BASES

LCO (continued)

The primary containment air locks are required to be OPERABLE. For each air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from primary containment.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining OPERABLE primary containment air locks in MODE 4 or 5 to ensure a control volume is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs) or during fuel movement of recently irradiated fuel assemblies in the primary containment. Due to radioactive decay, primary containment air locks are only required during fuel handling in the primary containment involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~7~~ days). 24 hours.

ACTIONS

The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door, then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the primary containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the primary containment boundary is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3 6.1.2.3

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure (Ref. 3), closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the nature of this interlock, and given that the interlock mechanism is only challenged when the primary containment airlock door is opened, this test is only required to be performed upon entering or exiting a primary containment air lock, but is not required more frequently than once per 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other administrative controls.

SR 3 6.1.2.4

A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > ~~1.28~~ 1.50 psig for a period of 24 hours from an initial pressure of 90 psig is an effective leakage rate test to verify system performance.

The 18 month Frequency is based on the fact that operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 3.8.
2. 10 CFR 50, Appendix J, Option B.
3. USAR, Table 6.2-1.
4. USAR, 15.7.4.
5. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

BASES

ACTIONS

F.1 and F.2 (continued)

vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. If suspending the OPDRVs would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valves to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR verifies that the 36 inch primary containment purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of the limits.

The SR is also modified by a Note (Note 1) stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. At times other than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation capability would be required by SR 3.6.1.3.4 and SR 3.6.1.3.7).

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The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for pressure control, ALARA, or air quality considerations for personnel entry or for Surveillances, or special testing on the purge system that require the valves to be open (e.g., testing of the containment purge radiation monitors). These primary containment purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other PCIV requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6 1.3.5

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 4), is required to ensure OPERABILITY. ~~The acceptance criterion for the purge supply valves is a combined leakage rate of ≤ 13,500 cc/hr for all required annulus bypass leakage paths when pressurized to ≥ P_a, 7.6 psig.~~ The acceptance criterion for each purge exhaust valve is ≤ 0.01 L_a when pressurized to ≥ P_a, 7.6 psig. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.6

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

evaluations of Reference 4 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated power operated or automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J, Option B maximum pathway leakage limits are to be quantified in accordance with Appendix J, Option B). The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 5).

A note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions the Reactor Coolant System is not pressurized and primary containment leakage limits are not required.

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage out of the primary containment that is less than the specified leakage rate. Leakage through the valves sealed in each division of MS-PLCS must be ≤ 150 scfh per division when tested at ≥ P_a, 7.6 psig. The leakage rate must be verified to be in accordance with the leakage test requirements of Reference 4, as modified by approved exemptions.

A note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required. The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 5).

(continued)

The leakage rate of 50 scfh when pressurized to ≥ Pa, 7.6 psig, per main steam line provides assurance that the assumptions in the radiological evaluations of Reference 4 are met.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

Surveillance of hydrostatically tested lines at ≥ 1.1 P_a, 8.36 psig provides assurance that the calculation assumptions of References 2 and 3 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 1.0 gpm times the total number of hydrostatically tested PCIVs when tested at 1.1 P_a. The combined leakage rates must be demonstrated at the frequency of the leakage test requirements of the Primary Containment Leakage Rate Testing Program (Ref. 5).

A note is added to this SR which states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

SR 3.6.1.3.12

This SR ensures that the combined leakage rate of annulus bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations of Reference 4 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated power operated or automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J, Option B maximum pathway leakage limits are to be quantified in accordance with Appendix J, Option B). The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

primary containment (Ref.2), to limit doses at the site boundary to within limits. The primary containment performs no active function in response to this event; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage rates assumed in safety analyses.

The FHA inside the primary containment is assumed to occur only after ≥ 80 hours since the reactor was last critical. The fission product release is, in turn, based on an assumed leakage rate from vent and drain valves with a combined flow rate of 70.2 cfm (based on an assumed 0.367 inch water gauge differential pressure). This assumed pressure reflects the fact that the FHA does not produce elevated containment pressures as is the case for the DBA LOCA. However, as an added conservatism, the analysis assumes a non-mechanistic additional leakage of 0.26% of the containment volume per day.

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by providing a contained volume to limit fission product escape following a FHA or other unanticipated reactivity or water level excursion. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Since no credit is assumed for automatic isolation valve closure, and any leakage which would occur prior to valve closure is similarly not accounted for, all penetrations which could communicate gaseous fission products to the environment must remain closed.

However, a limited number of primary containment penetration vent and drain valves may remain opened, and the primary containment considered OPERABLE provided the calculated leakage flow rate through the open vent and drain valves is less ≤ 70.2 cfm.

Leakage rates specified for the primary containment and air locks, addressed in LCO 3.6.1.1 and LCO 3.6.1.2 are not directly applicable during the shutdown conditions addressed in this LCO.

(continued)

BASES (continued)

APPLICABILITY

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining an OPERABLE primary containment in MODE 4 or 5 to ensure a control volume, is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary containment. Due to radioactive decay, the primary containment is only required to be OPERABLE during fuel handling involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 days). 24 hours

Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1.

ACTIONS

A.1 and A.2

In the event that primary containment is inoperable, action is required to immediately suspend activities that represent a potential for releasing radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.10.1

This SR verifies that each primary containment penetration that could communicate gaseous fission products to the environment during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Single isolation barriers that meet this criterion are a closed and de-activated power operated or automatic valve, a closed manual valve, a blind flange, or equivalent. This does not preclude the use of two active (ie, power operated and/or automatic) valves in the closed position for a given penetration. This SR does not require any testing or valve manipulation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.4 and SR 3.6.4.1.6

The SGT System exhausts the shield building annulus and auxiliary building atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the shield building annulus and auxiliary building that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the shield building annulus and auxiliary building to ≥ 0.5 and ≥ 0.25 inches of vacuum water gauge in ≤ 18.5 and ≤ (~~13.5~~ 34.5) seconds, respectively. This cannot be accomplished if the secondary containment boundary is not intact SR 3.6.4.1.6 demonstrates that each SGT subsystem can maintain ≥ 0.5 and ≥ 0.25 inches of vacuum water gauge for 1 hour. The 1 hour test period allows shield building annulus and auxiliary building to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure the integrity of this portion of the secondary containment boundary. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 15.6.5.

2. USAR, Section 15.7.4.

BASES

ACTIONS

A.1 (continued)

drywell is inoperable is minimal. Also, the Completion Time is the same as that applied to inoperability of the primary containment in LCO 3.6.1.1, "Primary Containment–Operating."

B.1 and B.2

If the drywell cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.1

The seal air flask pressure is verified to be at ≥ 75 psig every 7 days to ensure that the seal system remains viable. It must be checked because it could bleed down during or following access through the personnel door. The 7 day Frequency has been shown to be acceptable through operating experience and is considered adequate in view of the other indications available to operations personnel that the seal air flask pressure is low.

SR 3.6.5.1.2

A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > 0.67 psig for a period of 24 hours from an initial pressure of 75 psig is an effective leakage rate test to verify system performance. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.2.5 (continued)

system pressure does not decay at an unacceptable rate. The air lock seal will support drywell OPERABILITY down to a pneumatic pressure of 75 psig. Since the air lock seal air flask pressure is verified in SR 3.6.5.2.2 to be ≥ 75 psig, a decay rate ≤ 0.67 psig over 24 hours is acceptable. The 24 hour interval is based on engineering judgment, considering that there is no postulated DBA where the drywell is still pressurized 24 hours after the event. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage when the air lock OPERABILITY is not required. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix J.
2. USAR, Chapters 6 and 15.

B 3.7 PLANT SYSTEMS

B 3.7.2 Control Room Fresh Air (CRFA) System

BASES

BACKGROUND

The CRFA System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CRFA System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of recirculated air or outside supply air. Each subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork and dampers. Demisters remove water droplets from the airstream. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

In addition to the safety related standby emergency filtration function, parts of the CRFA System are operated to maintain the control room environment during normal operation. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CRFA System automatically switches to the isolation mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and control room air flow is recirculated and processed through either of the two filter subsystems. and 10 CFR 50.67

The CRFA System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA, per the requirements of GDC 19. CRFA System operation in maintaining the control room habitability is discussed in the USAR, Sections 6.4.1 and 9.4.1 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CRFA System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the USAR, Chapters 6

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and 15 (Refs. 3 and 4, respectively). The isolation mode of the CRFA System is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room.

The CRFA System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant subsystems of the CRFA System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in a failure to meet the dose requirements of GDC 19 in the event of a DBA. and 10 CFR 50.67

The CRFA System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, and 3, the CRFA System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRFA System

(continued)

BASES

APPLICABILITY (continued)

OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs); and

b. During the movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

A.1

With one CRFA subsystem inoperable, the inoperable CRFA subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRFA subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of CRFA System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CRFA subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

recently

fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the primary containment or fuel building or during OPDRVs, if the inoperable CRFA subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRFA subsystem may be placed in the emergency mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

recently

If applicable, movement of irradiated fuel assemblies in the primary containment or fuel building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both CRFA subsystems are inoperable in MODE 1, 2, or 3, the CRFA System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

recently

During movement of irradiated fuel assemblies in the primary containment or fuel building or during OPDRVs, with two CRFA subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, movement of irradiated fuel assemblies in the primary containment and fuel building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that a subsystem in a standby mode starts on demand from the control room and continues to operate with flow through the HEPA filters and charcoal adsorbers. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized to demonstrate the function of the system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 6.4.1.
2. USAR, Section 9.4.1.
3. USAR, Chapter 6.
4. USAR, Chapter 15.
5. Regulatory Guide 1.52, Revision 2, March 1978.
6. 10 CFR 50.67

BASES (continued)

LCO

Two independent and redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs) and;

b. During movement of recently irradiated fuel assemblies in the primary containment or fuel building.

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources–Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources–Operating."

recently

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary containment or fuel building ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

In general, when the unit is shut down the Technical Specifications (TS) requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence significantly reduced or eliminated, and minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for required systems.

During MODES 1, 2, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

BASES

LCO (continued)

nonessential loads, is a required function for DG OPERABILITY. In addition, proper load sequence operation is an integral part of offsite circuit and DG OPERABILITY since its inoperability impacts the ability to start and maintain energized any loads required OPERABLE by LCO 3.8.10.

It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required AC electrical power distribution subsystems.

As described in Applicable Safety Analyses, in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty.

recently

APPLICABILITY

The AC sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary containment or fuel building provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

recently

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

(continued)

BASES

ACTIONS (continued)

A.1

An offsite circuit is considered inoperable if it is not available to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.10, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required features inoperable which are not powered from offsite power, appropriate restrictions can be implemented in accordance with the required feature(s) LCOs' ACTIONS. Required features remaining powered from offsite power (even though that circuit may be inoperable due to failing to power other features) are not declared inoperable by this Required Action.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the primary containment or fuel building, and activities that could potentially result in inadvertent draining of the reactor vessel.

recently

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to initiate action immediately to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources– Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel building ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One DC electrical power subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division, associated with Division

(continued)

BASES

LCO (continued)

I or II onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems–Shutdown" is required to be OPERABLE. Similarly, when the High Pressure Core Spray (HPCS) system is required to be OPERABLE, the Division III DC electrical power subsystem associated with the Division III onsite Class 1E DC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.10 is required to be OPERABLE. In addition to the preceding subsystems required to be OPERABLE, a Class 1E battery or battery charger and the associated control equipment and interconnecting cabling capable of supplying power to the remaining Division I or II onsite Class 1E DC electrical power distribution subsystem(s), when portions of both Division I and II DC electrical power distribution subsystems are required to be OPERABLE by LCO 3.8.10. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel building provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

(continued)

BASES (continued)

recently

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.10, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated DC power source(s) inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and any activities that could result in inadvertent draining of the reactor vessel).

recently

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient accident analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to portions of the ESF instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

recently

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 4 and 5, and during movement of irradiated fuel assemblies in the primary containment or fuel building ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability are available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The inverters were previously identified as part of the Distribution System and, as such, satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABLE inverters require the associated AC vital bus be powered by the inverter through inverted DC voltage from the required Class 1E battery, or from an internal AC source via a rectifier with the battery available as backup. This ensures the availability of sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).

recently

APPLICABILITY

The inverters required to be OPERABLE in MODES 4 and 5 and also any time during movement of irradiated fuel assemblies in the primary containment or fuel building provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

recently

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

(continued)

BASES

ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two divisions are required by LCO 3.8.10, "Distribution Systems–Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required feature(s) to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required feature(s) inoperable with the associated inverter(s) inoperable, appropriate restrictions are implemented in accordance with the affected required feature(s) of the LCOs' ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the primary containment and fuel building, and any activities that could result in inadvertent draining of the reactor vessel). (recently)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, the unit is still without sufficient AC vital power sources to operate in a safe manner. Therefore, action must be initiated to restore the minimum required AC vital power sources and continue until the LCO requirements are restored.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems–Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems–Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

recently

The OPERABILITY of the minimum AC, DC, and AC vital bus electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary containment or fuel building ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

recently

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the primary containment and fuel building and any activities that could result in inadvertent draining of the reactor vessel). recently

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal–shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR-SDC inoperable, which results in taking the appropriate RHR-SDC ACTIONS.

(continued)

Attachment 4

Letter RBG-46052

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Deleted			

Attachment 5

Letter RBG-46052

POST-LOCA SUPPRESSION POOL pH EVALUATION SUMMARY

POST-LOCA SUPPRESSION POOL pH EVALUATION SUMMARY

SCOPE

The NUREG-1465 accident isotopic release specification allows deposition of iodine in the suppression pool. The iodine is assumed to remain in solution as long as the pool pH is maintained above 7. Upon detection of symptoms indicating that core damage is occurring, RBS procedures shift over to severe accident scenarios. Operators are directed to manually initiate the Standby Liquid Control System (SLCS) upon initiation of severe accident procedures. No credit is taken for any operator action during the first 10 minutes of an event. If an accident were to occur which would create the fuel damage conditions assumed in the analyses, it is reasonable to assume that manual initiation of SLCS injection would be initiated promptly. For the purposes of the analysis, however, this action is conservatively delayed by releasing the system's 1657 gallon inventory of sodium pentaborate solution into the RPV 2 hours after the postulated accident. The buffering effect of the SLCS solution is shown to maintain the suppression pool pH above 7 for the 30-day duration of the postulated LOCA and prevent iodine re-evolution.

INPUTS AND ASSUMPTIONS

The inputs and assumptions used in the evaluation of the post-LOCA suppression pool pH are listed in Tables 1. These inputs are consistent with the requirements of RG 1.183 and the approach in NUREG/CR-5950, "Iodine Evolution and pH Control." The analysis conservatively maximizes the acid contributions from Hydroiodic Acid from core halogens, Nitric Acid from radiolysis of water, and Hydrochloric Acid from radiolysis of chloride bearing cables inside containment while not crediting the core Cesium Hydroxide.

RESULTS

The results of the RBS post-LOCA suppression pool pH evaluation, assuming full injection, are provided in Table 2. The results show that the suppression pool pH will be maintained above 7.0 throughout the duration of the accident, thus, preventing re-evolution of elemental iodine dissolved in the pool water.

A parametric study was also performed. The results of the parametric study are presented in Figure 1. The study demonstrates that only 30% of the sodium pentaborate is required to reach the suppression pool to ensure that the pH remains above 7.0.

TABLE 1

RBS POST-LOCA SUPPRESSION POOL pH EVALUATION PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
1. Suppression Pool Volume	1,045,560 gallons
2. Minimum Initial pH value	5.3
3. Minimum SLCS injection rate	41.2 gpm
4. Minimum SLCS available volume	1657 gallons
5. Minimum sodium pentaborate weight percent in SLCS solution	7.13%
6. Start time for SLCS injection	Analysis assumes total inventory released at 2 hours. This value bounds initiation within the first hour and ~40 minutes of injection.
7. Suppression Pool Mixing	The suppression pool is assumed to be well mixed.
8. Core Thermal Power	3100 MWt
9. Core Activity Release	25% alkalis (5% gap, 20% early in vessel release) 30% iodines (5% gap, 25% early in vessel release)
10. Total Exposed Chloride Bearing Cable Mass	38007 lbm
11. HCl production from cables	Per Appendix B of NUREG/CR-5950. Beta dose to cables are based on current TID dose rates. Beta doses to cables in trays reduced by a factor of 2 per NUREG-0588, Section 1.4
12. CsOH	No credit for CsOH reaching the Suppression Pool

TABLE 2

RBS POST-LOCA SUPPRESSION POOL pH EVALUATION

Time(hrs)	HI (moles)	HNO_3 (moles)	HCl (moles)	pH
2	0.76	43.3	58.3	8.59
4	0.78	67.9	93.4	8.57
8	0.78	100.2	143.1	8.54
16	0.78	136.8	206.3	8.50
24	0.78	158.0	247.3	8.48
48	0.78	221.5	346.4	8.42
72	0.78	259.5	419.6	8.38
120	0.78	309.3	526.2	8.33
168	0.78	343.2	602.0	8.29
240	0.78	379.9	682.9	8.25
480	0.78	453.4	827.7	8.17
720	0.78	497.3	898.9	8.13

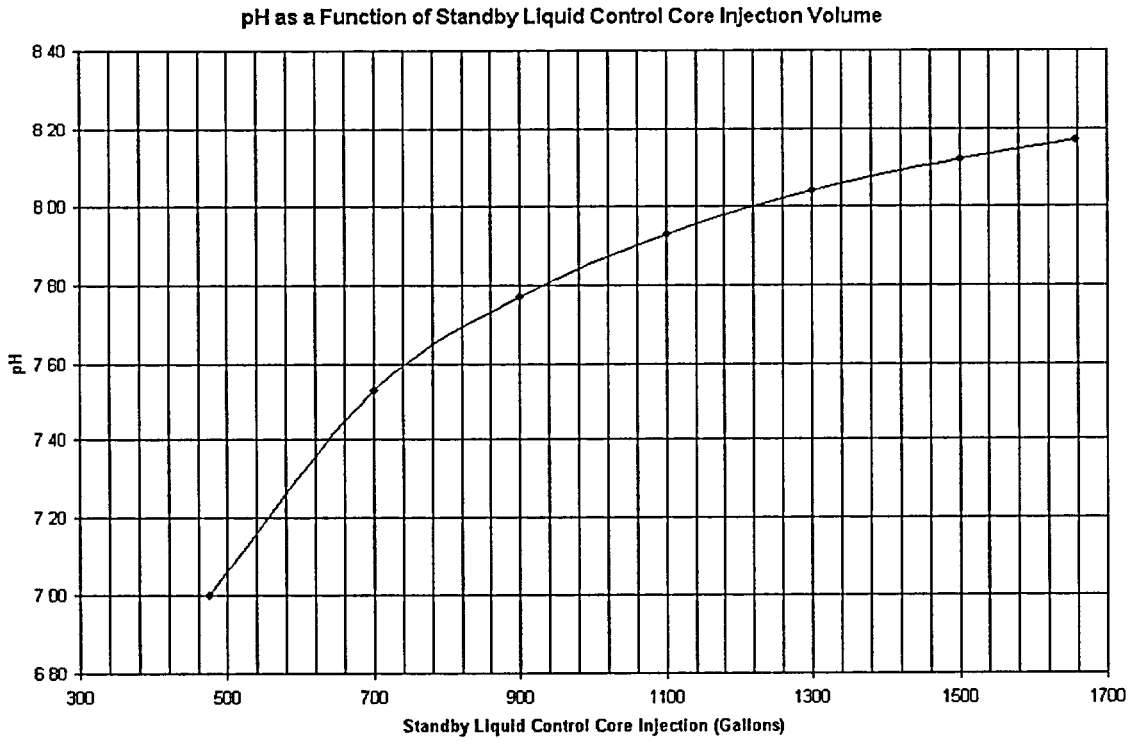


Figure 1

Attachment 6

Letter RBG-46052

Atmospheric Dispersion Factors (χ/Q) Calculations

Atmospheric Dispersion Factors (χ/Q) Calculations

In support of the Alternate Source Term project at RBS the Atmospheric Dispersion Factors (χ/Q) were reviewed. The χ/Q values currently used in RBS accident dose analyses are based on methodologies from Murphy and Campe's, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," for the Main Control Room and Regulatory Guide 1.145, Revision 1 for off-site locations. Also, the data for the χ/Q values currently used was taken between 3/17/77 and 3/16/79 (USAR Section 2.3). The χ/Qs were recalculated using current NRC codes and methodologies. Specifically, the NRC codes ARCON96 and PAVAN were used to recalculate the χ/Q values for the Main Control Room (MCR) and off-site locations, respectively. The meteorological data used (1/95 through 12/00) is more recent than that used in the original analysis.

Main Control Room Calculations

The Main Control Room χ/Q values were originally determined using the methodology found in Murphy and Campe and Regulatory Guide 1.145, Revision 1. The recalculated values were generated using the computer code ARCON96 as described in NUREG/CR-6331, *Atmospheric Relative Concentrations in Building Wakes,* Revision 1. A number of input assumptions are consistent with DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Table A-1. A summary of input data and assumptions may be found in Table 1.

Atmospheric stability was classified according to the temperature gradient values listed for the seven Pasquill stability categories in Regulatory Guide 1.23. Environmental data was obtained from the permanently installed meteorological tower and associated equipment. Joint wind speed, wind direction, atmospheric stability summaries, based on wind speed and wind direction at the 30-ft tower level, and the temperature difference between the 30-ft and 150-ft levels were obtained by the plant's permanently installed meteorological tower. Five years' worth of meteorological data was used in the evaluation. This data was taken from two separate time periods[1]: January 1, 1995 through December 31, 1998; and January 1, 2000 through December 31, 2000. Use of these dates provided over 41,400 hours of meteorological data for the 5 year period which exceeds the minimum requirement of 90%.

[1] The hourly average data required by ARCON96 could not be recovered for 1999 due to the failure of the tapes containing the data. This was not of concern for the off-site calculations since PAVAN uses the joint frequency tables which were available elsewhere. Two separate time periods were chosen to ensure that the most recent meteorological data was used in the evaluation.

The χ/Q values were calculated for both the main Control Room main air intake and the remote air intake. The main intake is located on the east side of the roof of the control building and the remote intake is located on the west side of the standby cooling tower. Six release points were considered. Each release path was modeled as a ground level point source with zero vent velocity and zero vent flow. The following release points were evaluated:

- the Main Plant Stack (Standby Gas Treatment System),
- the Main Steam Blowout Panel, and
- the Turbine Building.

The main plant stack release point is assumed for containment and Engineered Safety Features (ESF) liquid leakage in the LOCA dose analysis. For Secondary Containment Bypass leakage in the LOCA analysis the turbine building is the assumed release point. The Main Steam Line Break (MSLB) analyses release point is assumed to be the Main Steam Blowout Panel. The Control Rod Drop Accident (CRDA) assumes a release from the turbine building. The fuel handling accident used the previous χ/Q values for the Containment Building. These values were determined in accordance with the Murphy-Campe methodology. Additional χ/Q values were calculated for the Fuel Building and the Containment Equipment Hatch to ensure that the Murphy-Campe values were conservative.

The results of the analyses may be viewed in Table 2 below. Review indicates that the χ/Q calculated using ARCON96 and the 1995-2000 data yields much more favorable results that the previous evaluations. For example, the 0-2 hour Containment/SGTS χ/Q values were calculated to be roughly 50% and 20% of the current values for the main and remote intakes, respectively. Section 3.2 of NUREG/CR-6331, Revision 1 discusses the differences between the Murphy-Campe model and ARCON96 modeling. The NUREG states that "the Murphy-Campe model did not predict the variations of the concentrations in the vicinity of buildings particularly well. The studies also showed that one of the primary reasons that the Murphy-Campe modes did not predict concentration well was that it over-predicted concentrations during low speed conditions." The Murphy-Campe evaluations for RBS used a 1.3 mph (0.58 m/s) for the containment release for the 0-8 hour time period. Figure 27 from the NUREG shows Murphy-Campe/ARCON concentrations by wind speed. The figure implies that for 0.6 m/s the ratio can range between 20 to 200. The ratio between values calculated for RBS is on the order of 2 – 5. Further comparison between the Murphy-Campe and ARCON methodology indicated that several key assumptions were significantly different. For example, the Murphy-Campe evaluations assumed a diffuse release whereas the ARCON96 evaluations assumed a point source. Also, the occupancy factors for the MCR are inherent in the Murphy-Campe values. Therefore, based on this review it was concluded that the results calculated by ARCON96 were reasonable and appropriate for use at RBS.

Off-Site Calculations: Exclusion Area Boundary and Low Population Zone

Atmospheric dispersion factors were calculated for two locations, the exclusion area boundary (EAB) and the low population zone (LPZ). The off-site evaluations used the computer code PAVAN as described in NUREG/CR-2858, An Atmospheric Dispersion Program for Evaluation Design Basis Accidental Releases of Radioactive Materials from

Nuclear Power Stations." The joint frequency tables for six years of data (1994 through 1999) were used in this evaluation. These tables were generated utilizing data obtained from the site meteorological equipment discussed above. The bounding release points were evaluated:

- the Main Plant Stack (Standby Gas Treatment System),
- the Main Steam Blowout Panel, and
- the Turbine Building.

The release points for off-site doses are identical to the release points for the main control room doses. Note that the off-site doses for the FHA also utilize the previous χ/Q values. The containment release point was also explicitly evaluated to ensure that the previous χ/Q values were conservative.

The EAB is designated by a 3,000-ft (914 m) radius circle drawn about the reactor center. Although it is deemed "off-site," the EAB is entirely within RBS property, as shown in USAR Figure 2.1-2. The LPZ surrounding River Bend Station encompasses an area within a distance of 2.5 miles. The distance for the LPZ was chosen based on the requirements of 10CFR100.11. USAR Figure 2.1-8 shows roads and facilities within the LPZ.

Table 3 contains a summary of the meteorological data used in the off-site atmospheric dispersion factor analyses. The distances used in the analyses were based on the shortest distance from the release point to the EAB and LPZ. The values used are contained in Table 4 below. The top of the containment structure is 51.5 m above grade. The PAVAN results for the intermediate time steps were verified by hand calculations. The EAB values are found in Table 5, and the results of the LPZ calculations are presented in Table 6.

Table 1
Main Control Room Atmospheric Dispersion Factors - ARCON96 Inputs

Parameter	Main Plant Stack	Main Steam Tunnel Blowout Panel	Contain. Eq. Hatch	Turbine Building (worst point)		Fuel Handling Building	Comments
Lower Meas. Height, m	9.1	9.1	9.1	9.1		9.1	Lower instrument is 30' above grade.
Upper Meas. Height, m	45.7	45.7	45.7	45.7		45.7	Upper instrument is 150' above grade.
Wind Speed Units	m/s	m/s	m/s	m/s		m/s	
Release Height – MAI, m	58.8	22.5	2.5	28.3		22.9	
Release Height – RAI, m	58.8	22.5	2.5	28.3		3.0	
Building Area – MAI, m^2	2,121	1,006	2,121	909.5		838	
Building Area – RAI, m^2	2,121	911.5	2,121	911.5		838	
Vertical Velocity, m/s	0	0	0	0		0	Point releases – set to 0 per DG-1111, Table A-1
Stack Flow, m^3/s	0	0	0	0		0	Flow not credited – Set to 0 per DG-1111, Table A-1
Stack Radius, m	0	0	0	0		0	Set to 0 per DG-1111, Table A-1
Distance to Main Intake, m	61.9	61.7	56.7	42.8		67.4	See Figure 1 for release points (Note 1).
Distance to Remote Intake, m	118.2	151.2	119.7	136		72.7	See Figure 1 for release points (Note 1).
Main Intake Height, m	18.0	18.0	18.0	18.0		18.0	
Remote Intake Height, m	9.0	9.0	9.0	9.0		9.0	
Elevation Difference (MAI), m	0.0	0.0	0.0	1.1		1.1	
Elevation Difference (RAI), m	1.1	1.1	1.1	0.0		0.0	

Parameter	Main Plant Stack	Main Steam Tunnel Blowout Panel	Contain. Eq. Hatch	Turbine Building (worst point)		Fuel Handling Building	Comments
Direction to Source (MAI), °	255	214	274	177		281	
Direction to Source (RAI), °	099	111	090	127		091	
Surface Roughness Length, m	0.2	0.2	0.2	0.2		0.2	DG-1111, Table A-1
Wind Direction Window, °	90	90	90	90		90	DG-1111, Table A-1
Min. Wind Speed, m/s	0.5	0.5	0.5	0.5		0.5	DG-1111, Table A-1
Avg. Sector Width Constant	4.3	4.3	4.3	4.3		4.3	DG-1111, Table A-1
Initial Diffusion Coefficients, m	0	0	0	0		0	Point releases – set to 0 per DG-1111, Table A-1
Hours in Averages	Default	Default	Default	Default		Default	Default values used per DG-1111, Table A-1
Minimum Number of Hours	Default	Default	Default	Default		Default	Default values used per DG-1111, Table A-1

Note 1: The fuel building assumes releases through the FB ventilation system for the Main Air Intake and assumes a release through the truck bay doors for the remote air intake. Radwaste building releases are assumed through the building ventilation system.

Table 2
Control Room 5% Probability Level χ/Q Values (Sec/m^3)

Time Period	Standby Gas Treatment System		Main Steam Tunnel Blowout Panel		Containment Equipment Hatch	
	Main Air Intake	Remote Air Intake	Main Air Intake	Remote Air Intake	Main Air Intake	Remote Air Intake
0 to 2 hours	1.08E-03	4.29E-04	1.42E-03	2.65E-04	1.21E-03	3.44E-04
2 to 8 hours	7.65E-04	3.46E-04	1.08E-03	2.16E-04	7.46E-04	2.27E-04
8 to 24 hours	3.43E-04	1.37E-04	4.56E-04	9.19E-05	3.39E-04	9.62E-05
1 to 4 days	2.47E-04	1.18E-04	3.50E-04	6.67E-05	2.65E-04	7.75E-05
4 to 30 days	2.18E-04	8.59E-05	2.58E-04	4.85E-05	2.20E-04	5.78E-05

Time Period	Turbine Building Vent				Fuel Handling Building Door	
	Main Air Intake	Remote Air Intake			Main Air Intake	Remote Air Intake
0 to 2 hours	3.02E-03	3.69E-04			1.09E-03	8.56E-04
2 to 8 hours	2.47E-03	3.08E-04			6.83E-04	5.53E-04
8 to 24 hours	1.05E-03	1.30E-04			3.17E-04	2.41E-04
1 to 4 days	9.01E-04	1.07E-04			2.34E-04	1.89E-04
4 to 30 days	6.74E-04	7.15E-05			1.95E-04	1.42E-04

Table 3
Total Valid Hours in Stability Class

YEAR	CLASS A	CLASS B	CLASS C	CLASS D	CLASS E	CLASS F	CLASS G	TOTAL
1994	330	545	380	3180	2540	851	774	8600
1995	381	626	400	2920	2375	881	958	8541
1996	523	814	423	2757	2367	907	866	8657
1997	545	708	415	2679	2053	765	888	8053
1998	189	347	293	3922	2192	777	852	8572
1999	474	575	368	2849	2367	927	1116	8676

Table 4
Off-Site - Minimum Distances and Sector

LOCATION	SECTOR	EAB (m)	LPZ (m)
Turbine Building	S	759	3868
Radwaste Building	SSW	822	3931
Main Steam Tunnel	S, ESE, SE, SSE	871	3980
Edge of Containment	All	894	4003
SGTS (plant stack)	SSE	914	4023
Standby Cooling Tower	W	807	3916

Table 5
Exclusion Area Boundary χ/Q Results

	Wind Frequency Distribution Turbine Building						Wind Frequency Distribution Main Steam Tunnel		
Time Period	0.5% Max χ/Q	Sector/Dist	5% Site Limit				0.5% Max χ/Q	Sector/Dist	5% Site Limit
0-2 Hrs	7.51E-04	SW / 759	6.26E-04				6 33E-04	SW / 871	5.27E-04

Notes:
(1) Units for relative concentration, χ/Q, values are in seconds per cubic meter (sec/m^3)
(2) 0 5% χ/Q values represent the maximum selected from among all sector-dependent values (as indicated Distance units are in meters

	Wind Frequency Distribution SGTS (plant stack)			Wind Frequency Distribution Edge of Containment					
Time Period	0 5% Max χ/Q	Sector/Dist	5% Site Limit	0.5% Max χ/Q	Sector/Dist	5% Site Limit			
0-2 Hrs	6.05E-04	SW / 914	5.03E-04	6.18E-04	SW / 894	5.14E-04			

Notes:
(1) Units for relative concentration, χ/Q, values are in seconds per cubic meter (sec/m^3).
(2) 0 5% χ/Q values represent the maximum selected from among all sector-dependent values (as indicated. Distance units are in meters.

Table 6
Low Population Zone χ/Q Results

	Wind Frequency Distribution Turbine Building						Wind Frequency Distribution Main Steam Tunnel		
Time Period	0.5% Max χ/Q	Sector/Dist	5% Site Limit				0.5% Max χ/Q	Sector/Dist	5% Site Limit
0-8 Hrs	7.79E-05	SW / 3868	6.78E-05				7.57E-05	SW / 3980	6 65E-05
8-24 Hrs	5.23E-05	Same	4 64E-05				5.08E-05	Same	4.54E-05
1-4 Days	2.21E-05	Same	2.03E-05				2 13E-05	Same	1.98E-05
4-30 Days	6 40E-06	WSW	6 23E-06				6.24E-06	WSW / 3980	6.01E-06

Notes:

(1) Units for relative concentration, χ/Q, values are in seconds per cubic meter (sec/m^3)

(2) 0 5% χ/Q values represent the maximum selected from among all sector-dependent values (as indicated). Distance units are in meters.

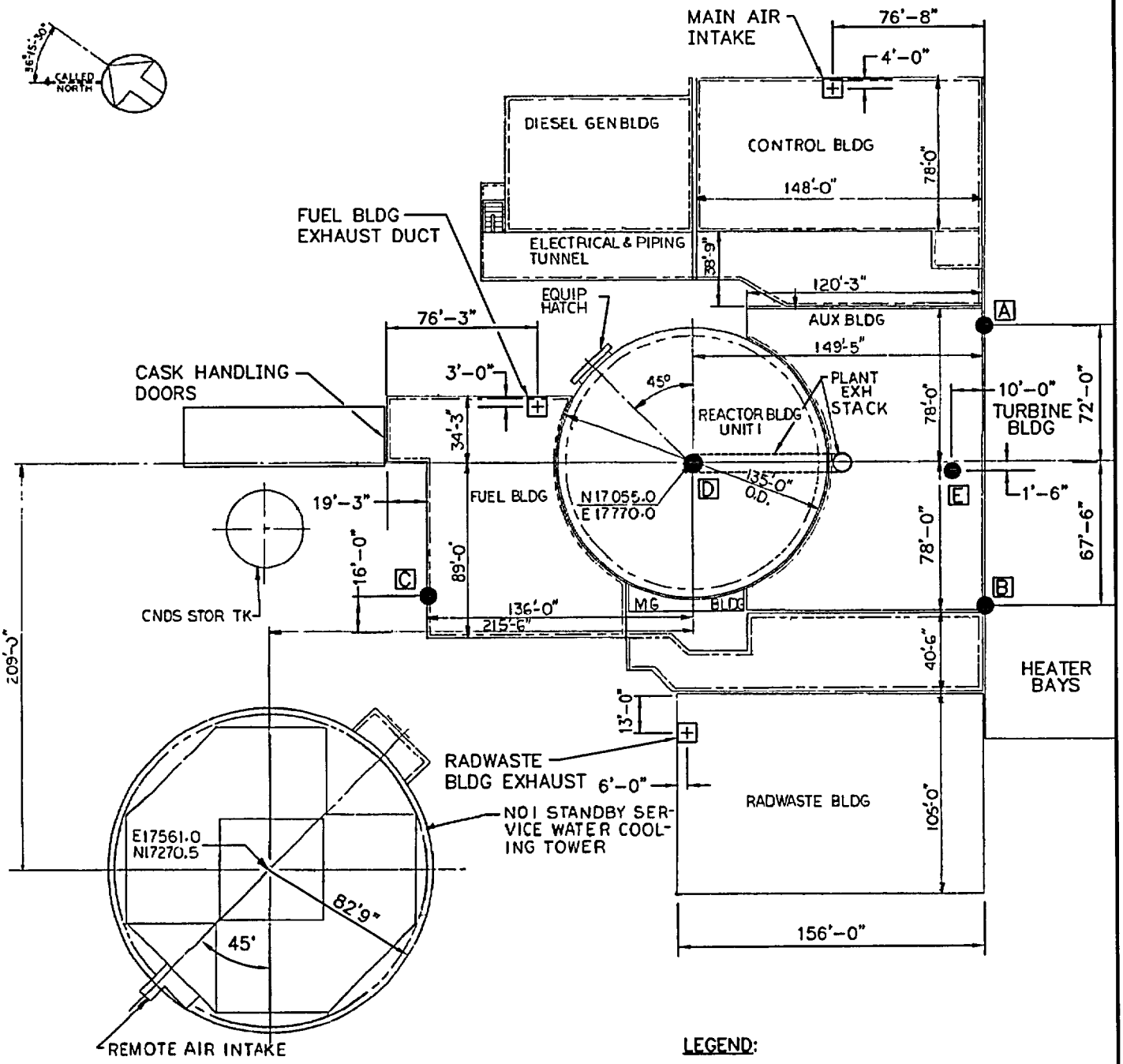
	Wind Frequency Distribution SGTS (plant stack)			Wind Frequency Distribution Edge of Containment					
Time Period	0 5% Max χ/Q	Sector/Dist	5% Site Limit	0.5% Max χ/Q	Sector/Dist	5% Site Limit			
0-8 Hrs	7.49E-05	SW / 4023	6 59E-05	7.53E-05	SW / 4003	6 62E-05			
8-24 Hrs	5.02E-05	Same	4 49E-05	5.04E-05	Same	4.51E-05			
1-4 Days	2.10E-05	Same	1.95E-05	2 11E-05	SW, WSW	1.96E-05			
4-30 Days	6 13E-06	WSW / 4023	5.91E-06	6.18E-06	WSW	5 95E-06			

Notes·

(1) Units for relative concentration, χ/Q, values are in seconds per cubic meter (sec/m^3).

(2) 0 5% χ/Q values represent the maximum selected from among all sector-dependent values (as indicated). Distance units are in meters

Figure 1
Main Control Room Atmospheric Dispersion Factor Release and Receptor Points



LEGEND:

- A TURBINE BUILDING RELEASE POINT FOR MAIN AIR INTAKE
- B TURBINE BUILDING RELEASE POINT FOR REMOTE AIR INTAKE
- C FUEL BUILDING RELEASE LOCATION FOR REMOTE AIR INTAKE
- D MAIN PLANT STACK (SGTS) RELEASE LOCATION
- E MAIN STEAM TUNNEL PRESSURE RELEASE PANELS

NOTE:
THIS FIGURE IS INTENDED TO SHOW GENERAL BUILDING LAYOUT

Entergy Operations, Inc.

Attachment 7

Letter RBG-46052

Loss of Coolant Accident (LOCA) Dose Analysis Summary

Loss of Coolant Accident (LOCA) Dose Analysis Summary

SCOPE

The Loss of Coolant Accident (LOCA) is postulated to occur as a consequence of a double ended guillotine break of a recirculation line. The LOCA is assumed to occur concurrently with a Safe Shutdown Earthquake (SSE), a Loss of Off-site Power (LOP), and a Single Active Failure (SAF) of an Emergency Diesel Generator (EDG). Additionally, a Main Steam Isolation Valve (MSIV) is assumed to fail to close which represents a second SAF. Traditionally only one SAF is required, however, for some doses the EDG failure would be the bounding assumption and for others the MSIV failure is likely to be bounding. Both SAF were assumed to prevent the necessity of performing detailed sensitivity analyses for each receptor location.

The computer code RADTRAD version 3.02 was used to determine the dose consequences in all of the analyses summarized in this submittal. The radiological consequences of the LOCA event are determined for the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and the Main Control Room (MCR). The calculated results are then evaluated against the acceptance criteria of 10CFR50.67.

INPUTS AND ASSUMPTIONS

The LOCA analysis evaluates four release points:

- Containment is assumed to leak at the proposed technical specification limit of 0.325 volume percent per day for the first 24 hours. The containment leakage rate (L_a) is reduced to 55% of that value at 24 hours based on the power uprate containment pressure response analysis as allowed per Regulatory Guide 1.183, Appendix A, Section 3.7. Since this term is in volume % per day, this leakage is assumed for both the drywell and containment. Manual initiation of the Standby Gas Treatment System (SGTS) is assumed, therefore, secondary containment releases were not credited for the first 30 minutes of the event. During this time period leakage from the primary containment is released directly to the environment. After 30 minutes containment leakage is directed to the annulus building which is treated by SGTS.

- The second contributor considered is Secondary Containment Bypass (SCB) leakage. SCB leakage is independent of L_a. SCB is assumed to leak at the proposed Technical Specification limit of 580,000 cc/hr (at P_a=7.6 psig) for the first 24 hours of the event. Since containment pressure is the driving force for this leakage term, SCB is also reduced to 55% of the original value after 24 hours based on the containment pressure response. This leakage is assumed to originate from the containment and is released directly to the environment via the turbine building for the duration of the event.

- The third contributor is 50 scfh through one main steam line in accordance with the proposed TS value. This leakage would only occur as a result of a failed MSIV. RBS currently has administrative limits of 30 scfh per MSIV and 10 scfh per drain line. The current TS limit is 150 scfh per division for all four main steam lines. The three remaining steam lines would not leak due to the fact that the pressure from the steam trapped between the MSIVs is significantly greater than the maximum containment pressure during an accident. This release is terminated 25 minutes into the event when the Main Steam Positive Leakage Control System (MSPLCS) becomes fully operational. This leakage is assumed to originate from the drywell and is released directly to the environment via the turbine building. Note that the current (i.e., TID) LOCA analysis neglects MSIV leakage in its entirety due to the trapped steam between the MSIVs and the fact that a failed MSIV is not assumed.

- The final contributor is the liquid leakage from Engineered Safety Features (ESF) cooling systems. Specifically, 1 gpm of suppression pool water is assumed to leak for the duration of the event. This leakage is assumed to be released directly to the environment for the first 30 minutes of the event. After 30 minutes the leakage is directed to the auxiliary building where it is treated by SGTS prior to release to the environment.

The fission product inventory used was based on that used in the power uprate submittal and is presented in Table 1. The release fractions assumed for each release phase are consistent with Regulatory Guide 1.183, Table 1. The release phases' start time and duration assumed are consistent with Regulatory Guide 1.183, Table 4. The dose conversion factors used are taken from Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," and FGR 12, "External Exposure to Radionuclides in Air, Water and Soil."

The activity released from the fuel is assumed to mix instantaneously with the drywell volume. Natural deposition of core halogens is credited in the drywell using the Powers (10%) model for BWRs from RADTRAD. Deposition for elemental iodine was also credited with a deposition coefficient of 1.01 hr^{-1}. RBS does not have containment sprays, therefore, none were credited in the analysis. Also, reduction of airborne radioactivity in the containment by suppression pool scrubbing was not credited.

The transport mechanisms between the drywell and containment vary as the event progresses. Early in the event the drywell has a significantly higher pressure than the containment due to the initial pipe break. However, roughly 10 minutes after the pipe break the drywell pressure drops below the containment pressure (See USAR Figure 6.2-5). The flow from the drywell to containment during this time period was calculated to be 4.74E+05 cfm based on an average drywell overpressure of 20 psid (from 2 – 10 minutes) and a flow area ($A/\sqrt{k}$) of 1.0 ft^2. After the drywell becomes "negative" no flow was modeled. Experience with the RBS model indicates that calculated doses are not sensitive to minor changes in this flow rate. The fuel damage postulated in the analysis requires a significant amount of core damage. This would also generate a significant amount of hydrogen from water-cladding reactions. The hydrogen mixing system is assumed to be initiated 25 minutes into the event (same start time as MS-PLCS). This would equalize the drywell and containment atmospheres. The flow rate for the hydrogen mixing system is 600 cfm. An additional 3000 cfm is assumed from the drywell to the containment to account for core steaming. This value is identical to that used by Grand Gulf and Perry which should bound RBS since both those BWR/6 plants are larger than RBS. Finally, just before 2

hours a very high mixing rate between the drywell and containment is assumed. Regulatory Guide 1.183, Appendix A, Section 3.7 states "After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment."

The Control Room Fresh Air (CRFA) ESF charcoal filters are credited in the LOCA dose analysis. The system is assumed to operate in normal mode for the first 20 minutes of the event, at which time manual initiation of the system is assumed to occur. These assumptions conservatively bound automatic initiation of the system. Breathing rates and occupancy factors are based on Regulatory Guide 1.183 guidance. The normal air intake is 2,000 cfm, however, an unfiltered inleakage of 300 cfm is conservatively assumed so the corrected flow intake flow rate is 1,700 cfm. A schematic of the actual and simulated CRFA system configuration is presented in Figures 4a and 4b for the normal and emergency modes, respectively. RBS has dual manual air intakes which meet the criteria set forth in Murphy and Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19." SRP 6.4.III.4.ii states that plants with dual inlet designs limited to manual selection may use the more favorable inlet location. Further the SRP allows the more favorable value to be reduced by a factor of 4. This credit will be taken for the normal air intake. The main air intake for the control room is located on the control building just above the control room envelope. The main air intake (MAI) χ/Qs will be assumed to apply to the unfiltered inleakage since the MAI is located on the roof of the Control Building just above the MCR envelope. The LOCA analysis utilized a flow weighted average of the applicable χ/Q values. These values are presented in Table 2.

$$\chi/Q_{effective} = \frac{1700\text{cfm}\left(\frac{\chi/Q_{most-fav}}{4}\right) + 300\text{cfm}\left(\chi/Q_{MAI}\right)}{2000\text{cfm}}$$

where,

$\chi/Q_{most-fav}$ =	More favorable χ/Q value (main air intake or remote air intake),
χ/Q_{MAI} =	χ/Q value corresponding to the main air intake, and
χ/Q_{eff} =	The effective χ/Q value used in the LOCA analysis.

Three separate RADTRAD input decks were required. One analyzed the dose consequences of primary containment leakage from containment and secondary containment. The postulated accident activity is released as a "ground level release" from the main plant stack (since the stack is not 2.5x the height of containment as would be required for a "stack" release). The second file evaluated the doses from MSIV and SCB leakage. Both of these leakage terms are assumed to originate from the Turbine Building. The third and final file evaluated the liquid ESF leakage which is also assumed to be released via the main plant stack.

RESULTS

The radiological consequences for the postulated LOCA event are summarized in Table 4. The LOCA event results in offsite and control room doses within the regulatory limits of 10CFR50.67. One exception to Regulatory Guide 1.183 guidance was taken:

- **Engineered Safety Features Liquid Leakage:** RG 1.183, Appendix A, Section 3.8 states that ESF leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications would require declaring such systems inoperable. The leakage rate assumed in this analysis is 1 gpm based on the initial licensing of the plant. Specifically, the RBS SER (NUREG-0989), Section 15.6.5, states that "The leak rate chosen for ESF equipment is that used by the staff in past evaluations; it represents the largest flow that could plausibly exist over the duration of the accident without discovery and isolation of this source." In addition to the initial licensing of the plant, this value was used in the Technical Specification Amendment 98, 113, and 114 submittals which were approved by the NRC.

To address potential MCR habitability issues a sensitivity study was performed. It was determined that the 5 REM (0.05 Sv) dose limit prescribed by 10CFR50.67 with unfiltered inleakage rates a high as 420 cfm. This analysis took no credit for dilution in other volumes of the control building (which is a safety related and seismically qualified structure) prior to leaking into the MCR.

It should be noted that the Vital Area Access (VAA) were reviewed using AST assumptions. The current values documented in the USAR were found to be bounding. Equipment Qualification (EQ) doses were not reevaluated as Regulatory Guide 1.183, Section 6, states that "licensees may continue to use either the AST or TID-14844 assumptions to perform the required EQ analyses." Design changes are evaluated by the River Bend EQ Program to ensure that all equipment will operate as required during postulated accidents.

Table 1

BWR CORE INVENTORIES

AST Group	Isotope	RADTRAD Inventory (Ci/MWt)	Power Uprate Inventory (Ci/MWt)	Assumed Inventory (Ci/MWt)
1	Kr-85	2.51E+02	3.02E+02	3.02E+02
1	Kr-85m	9.11E+03	6.73E+03	6.73E+03
1	Kr-87	1.66E+04	1.29E+04	1.29E+04
1	Kr-88	2.24E+04	1.83E+04	1.83E+04
1	Xe-133	5.43E+04	5.53E+04	5.53E+04
1	Xe-135	1.29E+04	7.15E+03	7.15E+03
2	I-131	2.58E+04	2.63E+04	2.63E+04
2	I-132	3.79E+04	3.85E+04	3.85E+04
2	I-133	5.42E+04	5.50E+04	5.50E+04
2	I-134	5.93E+04	6.06E+04	6.06E+04
2	I-135	5.10E+04	5.19E+04	5.19E+04
3	Rb-86	1.40E+01	4.70E+01	4.70E+01
3	Cs-134	4.23E+03	5.36E+03	5.36E+03
3	Cs-136	1.13E+03	1.18E+03	1.18E+03
3	Cs-137	2.53E+03	3.32E+03	3.32E+03
4	Sb-127	2.32E+03	2.28E+03	2.28E+03
4	Sb-129	8.07E+03	8.08E+03	8.08E+03
4	Te-127	2.25E+03	2.25E+03	2.25E+03
4	Te-127m	3.03E+02	3.41E+02	3.41E+02
4	Te-129	7.57E+03	7.60E+03	7.60E+03
4	Te-129m	1.99E+03	2.06E+03	2.06E+03
4	Te-131m	3.82E+03	3.73E+03	3.73E+03
4	Te-132	3.74E+04	3.79E+04	3.79E+04
5	Sr-89	2.77E+04	2.47E+04	2.47E+04
5	Sr-90	1.96E+03	2.58E+03	2.58E+03
5	Sr-91	3.60E+04	3.16E+04	3.16E+04
5	Sr-92	3.77E+04	3.37E+04	3.37E+04
6	Ba-139	4.99E+04	4.93E+04	4.93E+04
6	Ba-140	4.93E+04	4.75E+04	4.75E+04

TABLE 1 – cont.

BWR CORE INVENTORIES

AST Group	Isotope	RADTRAD Inventory (Ci/MWt)	Power Uprate Inventory (Ci/MWt)	Assumed Inventory (Ci/MWt)
7	Co-58	1.53E+02	Not listed	1.53E+02
7	Co-60	1.83E+02	Not listed	1.83E+02
7	Mo-99	4.86E+04	5.01E+04	5.01E+04
7	Tc-99m	4.20E+04	4.32E+04	4.32E+04
7	Ru-103	3.69E+04	4.24E+04	4.24E+04
7	Ru-105	2.46E+04	2.99E+04	2.99E+04
7	Ru-106	1.00E+04	1.51E+04	1.51E+04
7	Rh-105	1.84E+04	2.52E+04	2.52E+04
8	Ce-141	4.47E+04	4.40E+04	4.40E+04
8	Ce-143	4.36E+04	4.15E+04	4.15E+04
8	Ce-144	2.90E+04	3.53E+04	3.53E+04
8	Np-239	5.68E+05	Not listed	5.68E+05
8	Pu-238	3.95E+01	Not listed	3.95E+01
8	Pu-239	1.00E+01	Not listed	1.00E+01
8	Pu-240	1.25E+01	Not listed	1.25E+01
8	Pu-241	2.16E+03	Not listed	2.16E+03
9	Y-90	2.10E+03	2.79E+03	2.79E+03
9	Y-91	3.39E+04	3.22E+04	3.22E+04
9	Y-92	3.78E+04	3.39E+04	3.39E+04
9	Y-93	4.30E+04	3.91E+04	3.91E+04
9	Zr-95	4.46E+04	4.42E+04	4.42E+04
9	Zr-97	4.59E+04	4.54E+04	4.54E+04
9	Nb-95	4.22E+04	4.42E+04	4.42E+04
9	La-140	5.03E+04	5.03E+04	5.03E+04
9	La-141	4.64E+04	4.44E+04	4.44E+04
9	La-142	4.47E+04	4.34E+04	4.34E+04
9	Pr-143	4.26E+04	4.11E+04	4.11E+04
9	Nd-147	1.91E+04	1.81E+04	1.81E+04
9	Am-241	2.19E+00	Not listed	2.19E+00
9	Cm-242	5.79E+02	Not listed	5.79E+02
9	Cm-244	3.13E+01	Not listed	3.13E+01

Note: RADTRAD inventory values were used for isotopes which were not listed in the power uprate source term (Co, Pu, Am, and Cm isotopes).

TABLE 2

MAIN CONTROL ROOM FLOW BIASED X/Q VALUES

Time Period	Main Air Intake	Remote Air Intake	More favorable	MF/4	Effective
Main Plant Stack					
• 0 to 2 hours	1.08E-03	4.29E-04	4.29E-04	1.07E-04	2.53E-04
• 2 to 8 hours	7.65E-04	3.46E-04	3.46E-04	8.65E-05	1.88E-04
• 8 to 24 hours	3.43E-04	1.37E-04	1.37E-04	3.43E-05	8.06E-05
• 1 to 4 days	2.47E-04	1.18E-04	1.18E-04	2.95E-05	6.21E-05
• 4 to 30 days	2.18E-04	8.59E-05	8.59E-05	2.15E-05	5.10E-05
Turbine Building					
• 0 to 2 hours	3.02E-03	3.69E-04	3.69E-04	9.23E-05	5.31E-04
• 2 to 8 hours	2.47E-03	3.08E-04	3.08E-04	7.70E-05	4.36E-04
• 8 to 24 hours	1.05E-03	1.30E-04	1.30E-04	3.25E-05	1.85E-04
• 1 to 4 days	9.01E-04	1.07E-04	1.07E-04	2.68E-05	1.58E-04
• 4 to 30 days	6.74E-04	7.15E-05	7.15E-05	1.79E-05	1.16E-04

TABLE 3

X/Q VALUES USED IN LOCA ANALYSIS

Release Point	EAB*	LPZ	MCR
SGTS/Containment			
• 0-2 hours	6.05E-4	7.49E-5	2.53E-4
• 2-8 hours	6.05E-4	7.49E-5	1.88E-4
• 8-24 hours	6.05E-4	5.02E-5	8.06E-5
• 1-4 days	6.05E-4	2.10E-5	6.21E-5
• 4-30 days	6.05E-4	6.13E-6	5.10E-5
Turbine Building			
• 0-2 hours	7.51E-4	7.79E-5	5.31E-4
• 2-8 hours	7.51E-4	7.79E-5	4.36E-4
• 8-24 hours	7.51E-4	5.23E-5	1.85E-4
• 1-4 days	7.51E-4	2.21E-5	1.58E-4
• 4-30 days	7.51E-4	6.40E-6	1.16E-4

Note *: The 0 – 2 hour values was conservatively applies for the duration of the accident to ensure the "maximum" 2 hour dose is calculated as required per RG 1.183.

TABLE 4

RBS LOCA RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
I. Data and assumptions used to estimate radioactive source from postulated accident.	
1. Power Level	3100 MWt
2. Core Activity available for release	Table 1
3. Gap Activity Release Fractions	Per Table 1 of RG 1.183
4. Release fission product species and chemical form	Per RG 1.183, Section 3 5
II. Release Rates	
1. Primary Containment Leakage Rate • 0 – 24 hours • 1 – 30 days	 0.325 volume % per day 0.179 volume % per day
2. Secondary Containment Bypass Leakage Rate • 0 – 24 hours • 1 – 30 days	 580,000 cc/hr @ Pa (0.341 cfm) 319,000 cc/hr @ Pa (0.188 cfm)
3. Main Steam Line Leakage • 0 – 25 minutes • 25 minutes – 30 days Note *: 50 scfh was converted based on a maximum DW temperature of 330°F. The pressure assumed was 7.6 psig (power uprate reports show that the drywell pressure is 22.8 psia @ 121 seconds decreases steadily to ~19 psia at 10 minutes)	 50 scfh = 0.85cfm* 0 scfh
4. Engineered Safety Features Leakage	1 gpm
III. Dispersion Data	
1. EAB X/Q Data	See Table 3
2. LPZ X/Q Data	See Table 3
3. Control Room X/Q Data	See Table 3
IV. Control Room Parameters	
1. Unfiltered In-Leakage Rate	300 cfm
2. Outside Air Ventilation Rate • Actual • Assumed	 2000 cfm 2000 – 300 = 1700 cfm
3. Filter Initiation Time	20 minutes
4. CR ESF Iodine Filter Efficiency • Elemental/Organic (Charcoal) • Particulate (HEPA)	 98% 99%
5. Control Room Breathing Rates and Occupancy Factors	Per RG 1.183

Description of Input/Assumption	Design Basis Input and/or Assumption
V. Standby Gas Treatment Parameters	
1. Positive Pressure Period	30 minutes
2. Flow Rates • Annulus • Auxiliary Building	 2,500 cfm 10,000 cfm
3. SGTS Iodine Filter Efficiency • Elemental/Organic (Charcoal) • Particulate (HEPA)	 90% 99%
VI. Building Volumes	
1. Drywell	2.36E+05 ft^3
2. Containment	1.19E+06 ft^3
3. Annulus*	3.57E+05 ft^3
4. Auxiliary Building* Note *: Only 50% of the annulus and auxiliary building volumes were credited in the actual analysis (values listed are actual volumes)	1.16E+06 ft^3
5. Control Room	1.88E+05 ft^3
6. Suppression Pool** Note **: The actual analysis conservatively assumed a volume of 120,000 ft^3.	1.25E+05 ft^3
VII. Containment Mixing Data	
1. Blowdown Data (Drywell ⇒ Containment) • 0 – 10 minutes • 10 minutes +	 4.74E+05 cfm 0 cfm
2. Hydrogen Mixing Data (Drywell ⇔ Containment) • 0 – 25 minutes • 25 minutes – 1.9 hours • 1.9 hours – 30 days (Infinite Mixing)	 0 cfm 600 cfm 1.0E+08 cfm
3. Steaming Data (Drywell ⇒ Containment) • 0 – 25 minutes • 25 minutes – 1.9 hours • 1.9 hours – 30 days (Infinite Mixing)	 0 cfm 3000 cfm (Included in Hydrogen Mixing)
VIII. Misc. Data	
1. Dose Conversion Factors	Based on FGR 11 & 12
2. Off-Site Breathing Rates	Based on RG 1.183, Section 4.1.3
3. Drywell Plateout Coefficients • Elemental • Particulate	 1.01 hr^{-1} RADTRAD default - Powers (10) Model
4. ESF Leakage – Halogen Flashing Fraction	0.10

TABLE 5

AST SCENARIO TIMING

Time	AST Time	Event
0 sec	-121 sec.	• DER of one of the two Recirculation Lines in the reactor occurs. • Plant experiences a SSE. • Loss of Offsite Power occurs. • Release of all reactor coolant halogens in AST scenario.
0+ sec.	-121 sec.	• LOCA signal from high drywell pressure occurs effectively instantaneously. • Reactor low water level reached. • All Emergency Diesel Generators (EDG) signaled to start. • ECCS systems signaled to start. • Main Control Room Ventilation Signaled to start.
1.1 sec.*	-119.9 sec.	Drywell peak pressure reached.
5.5 sec.*	-114.5 sec.	7 of 8 MSIVs are closed (one MSIV assumed to fail open)
10 sec.*	-111 sec.	2 of 3 EDG start (1 assumed to fail to start) and ready to load.
23 sec.*	-98 sec.	Top of Active Fuel Uncovered (USAR Figure 6.3-11)
27 sec.*	-94 sec.	Initiation of HPCS.
37 sec. *	-84 sec.	Initiation of LPCS and LPSI.
56 sec.*	-65 sec.	Low Pressure Core Spray flow begins (initiating ESF leakage)
121 sec.	0 sec.	Failure of fuel cladding begins in AST scenario. This results in the release of gap activity. The following is assumed based on the above: • Containment leaking at proposed L_a • ESF leakage at 1 gpm • SCB leakage at proposed TS limit.
293 sec.*	172 sec.	Vessel is reflooded (Not in AST scenario which has no cooling).
584 sec.	463 sec.	Drywell pressure < Containment pressure. Suppression pool bypass flow assumed to terminate.
20 min.	18 min.	• Operators assumed to initiate the Main Control Room Fresh Air Emergency Filters. • Operators assumed to initiate Main Steam Positive Leakage Control System. • Operators assumed to initiate the SGTS system.
25 min.	23 min.	• MS-PLCS becomes fully operational terminating failed MSIV release. • Operators initiate the hydrogen mixing system.
30 min.	28 min.	Annulus and Auxiliary Building assumed to reach –0.25" w.g., therefore, secondary containment is established.
32 min.	30 min.**	Core geometry is compromised in AST scenario – fuel melting begins. Early In-Vessel Release begins.
<1 hour	<1 hour	Operators initiate Standby Liquid Control System.

Time	AST Time	Event
2 hours	2 hours	Fuel melting terminated.
8 hours	8 hours	MCR and LPZ X/Q changed.
24 hours	24 hours	• MCR and LPZ X/Q changed. • Off-site breathing rate reduced. • Containment and SCB leakage reduced to 55% of proposed TS allowable value based on containment pressure analyses. • MCR occupancy reduced.
96 hours	96 hours	• MCR and LPZ X/Q changed. • Off-site breathing rate reduced. • MCR occupancy reduced.
30 days	30 days	End of Dose Calculation.

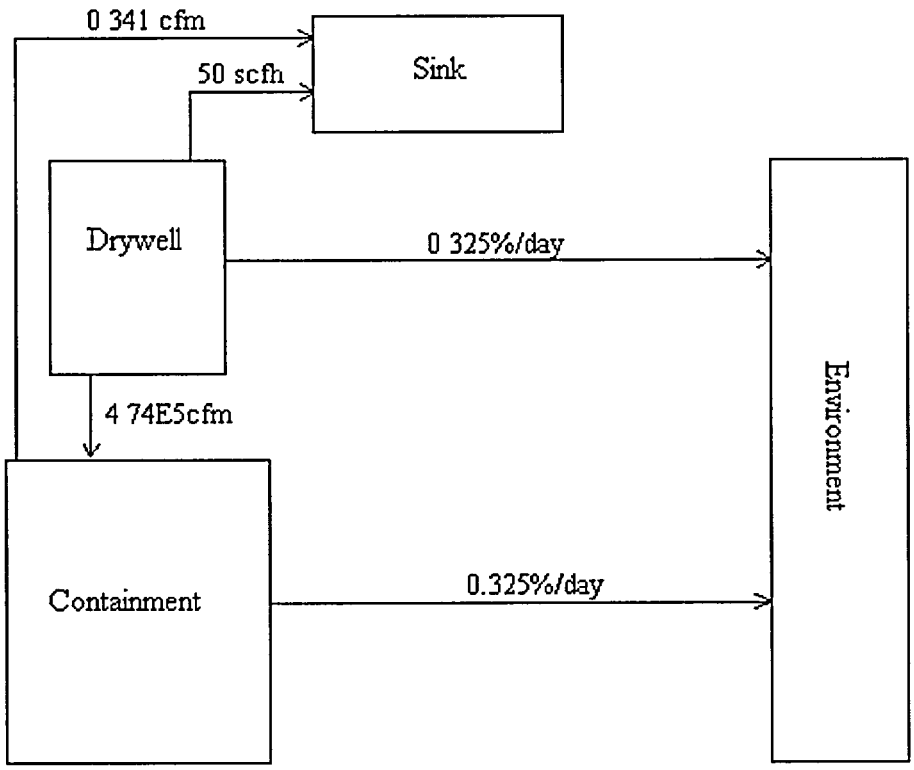
Note *: Minor changes in these times will not have an appreciable affect the dose calculation.
Note **: Times >30 minutes were no longer adjusted to account for the 2 minutes for fuel damage.

TABLE 6

LOCA RADIOLOGICAL CONSEQUENCES

Release Descriptions	EAB	LPZ	MCR
Containment/Secondary Containment	2.6	1.7	0.4
Secondary Containment Bypass/MSIV	12.3	5.4	2.9
ESF Liquid Leakage	<0.1	0.4	<0.1
Total	**15.0**	**7.5**	**3.4**
Regulatory Limit	***25.0***	***25.0***	***5.0***

Figure 1a
Primary and Secondary Containment Releases: 0 – 10 Minutes



Note *: The times referred to in Figures 1 through 3 are post-break times rather that AST time (i.e., time after the onset of fuel damage). This was done for convenience and to avoid confusion. In reality the input decks generated by RBS were done in AST time and neglect the first 2 minutes.
Note **: The "sink" node was used to contain radioactivity which is evaluated elsewhere. This was necessary to apply the appropriate dispersion factors to different release paths. Specifically, for the primary and secondary containment releases the SCB and MSIV leakage paths are evaluated in a separate input deck (as depicted in Figures 2a through 2e).

Figure 1b
Primary and Secondary Containment Releases: 10 – 25 Minutes

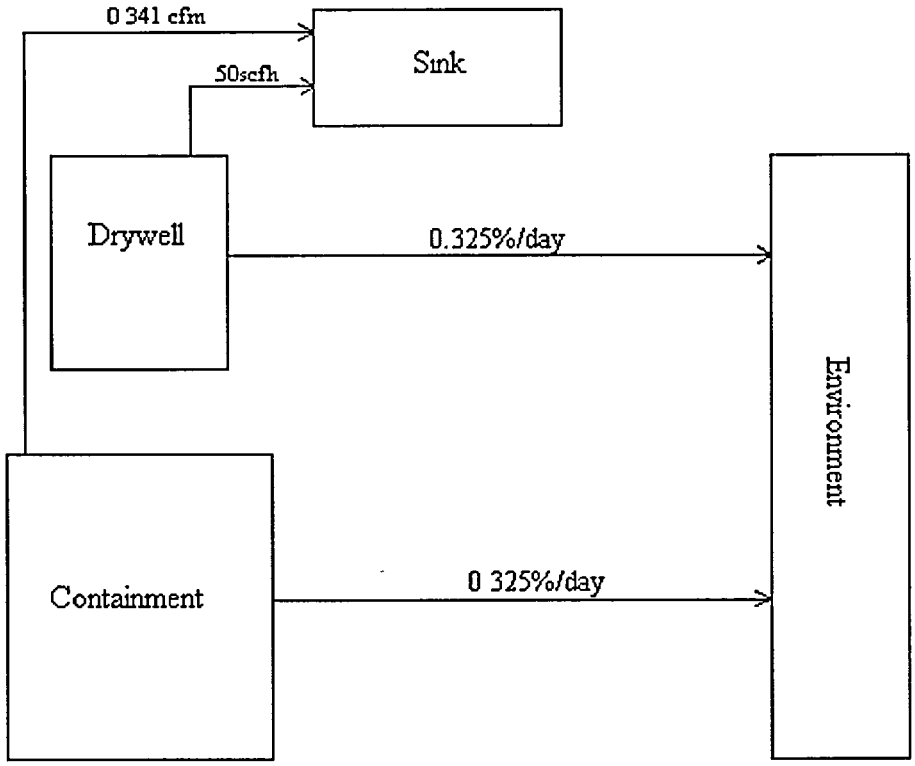


Figure 1c
Primary and Secondary Containment Releases: 25 – 30 Minutes

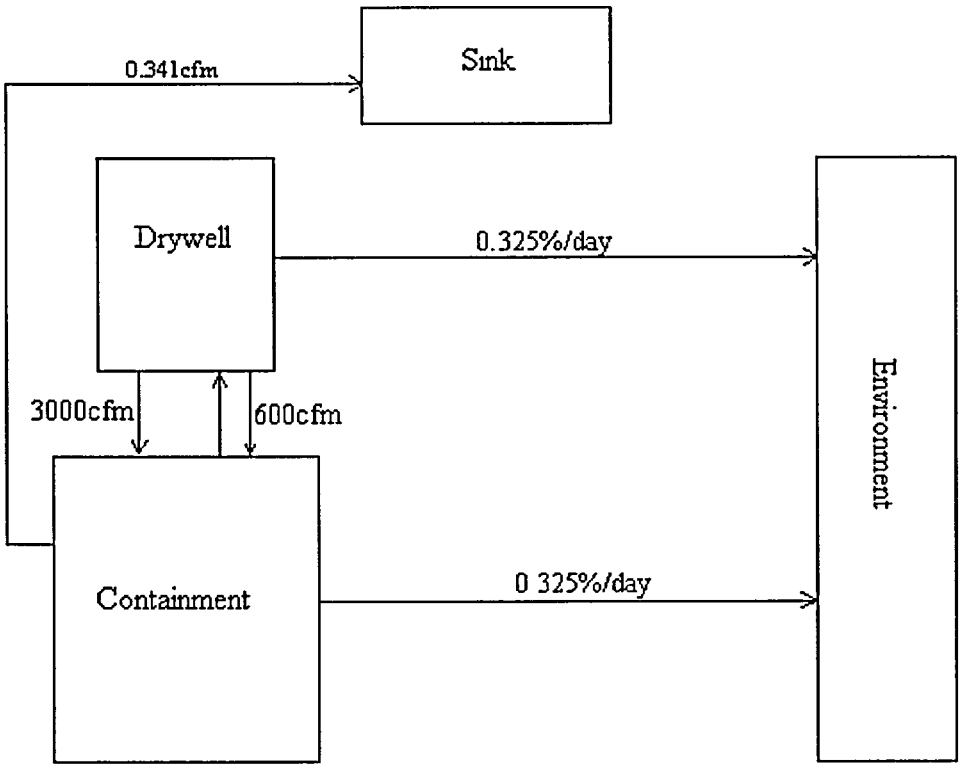


Figure 1d
Primary and Secondary Containment Releases: 30 Minutes - 1.9 Hours

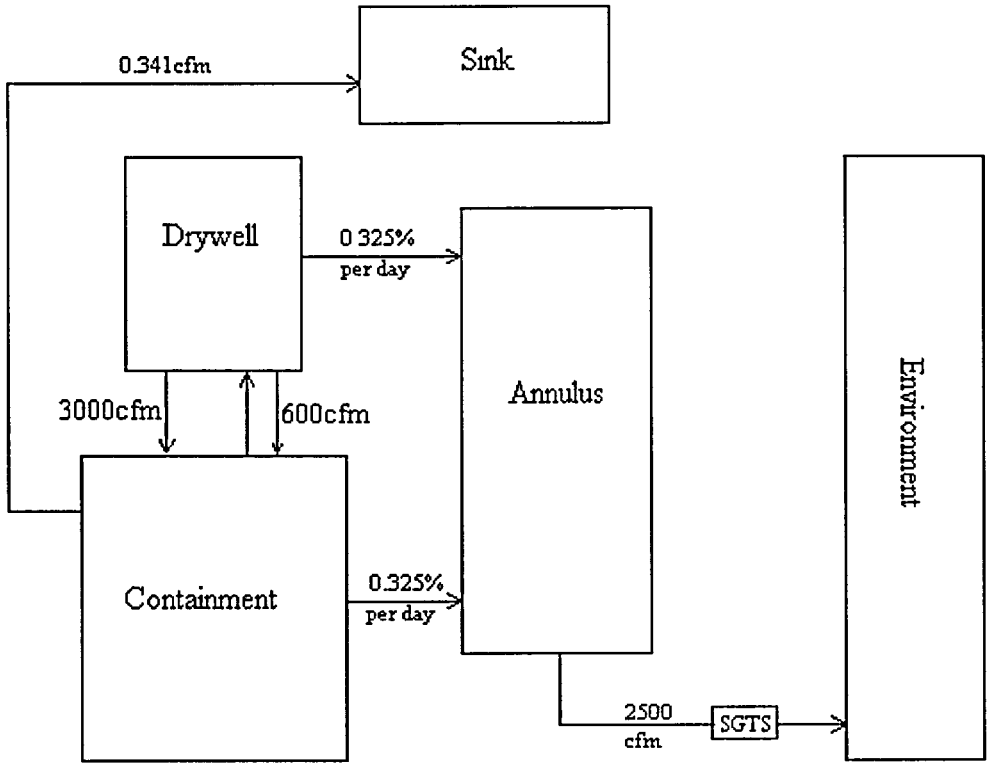


Figure 1e
Primary and Secondary Containment Releases: 1.9 – 24 Hours

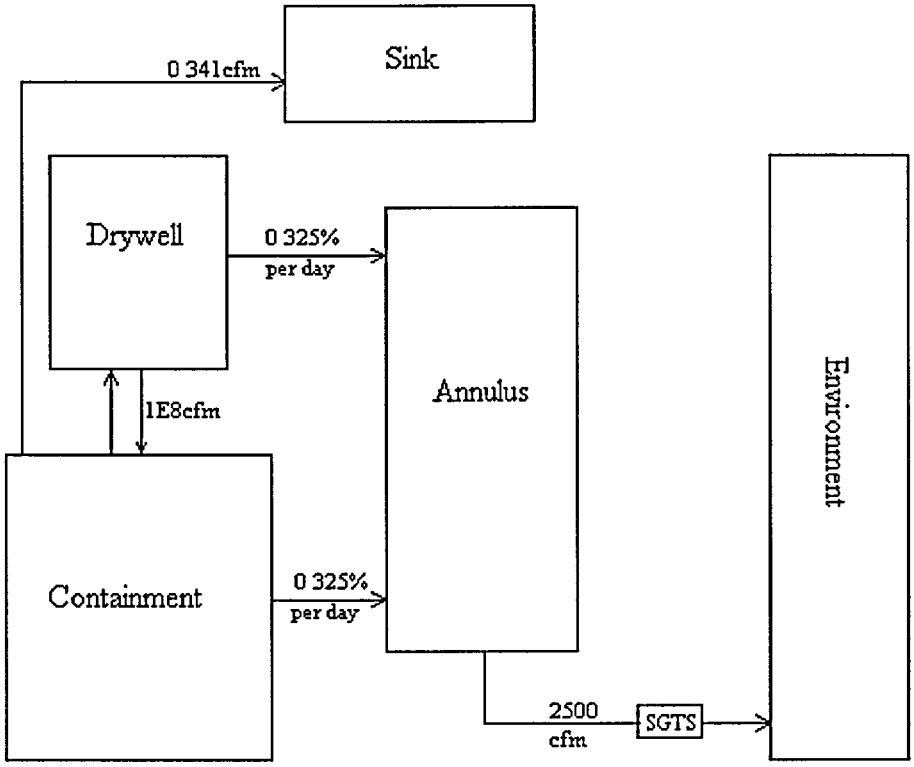


Figure 1f
Primary and Secondary Containment Releases: 1 – 30 Days

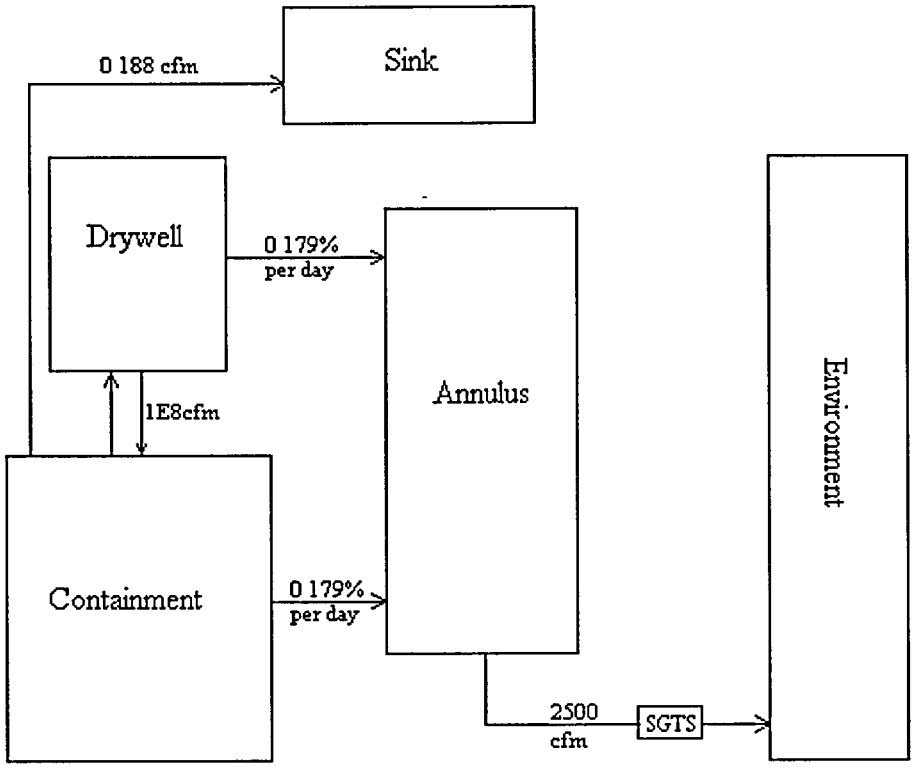


Figure 2a
Secondary Containment Bypass and MSIV Releases: 0 – 10 Minutes

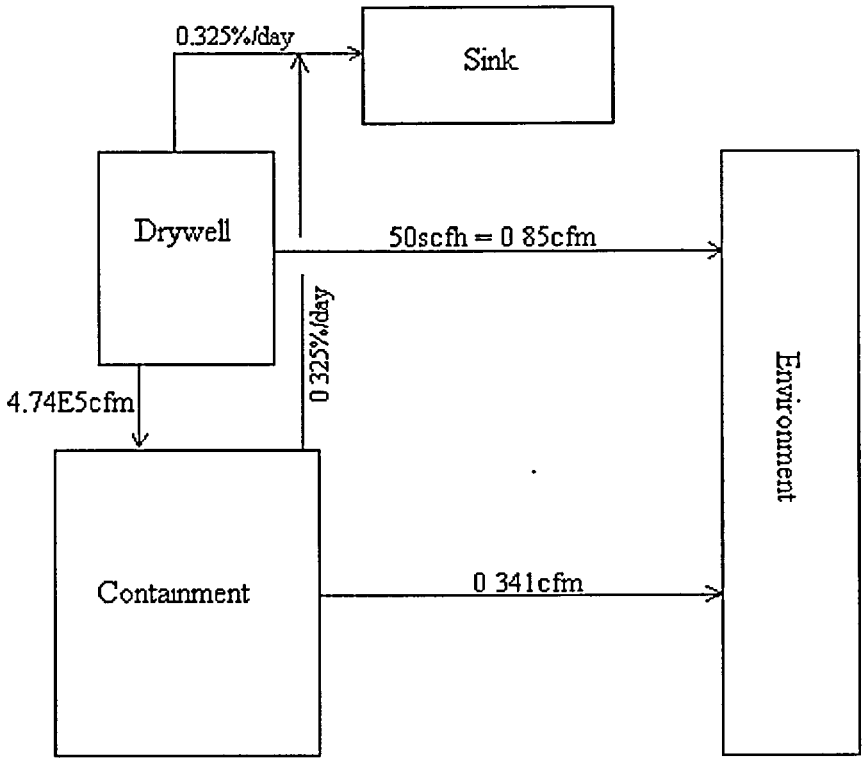


Figure 2b
Secondary Containment Bypass and MSIV Releases: 10 – 25 Minutes

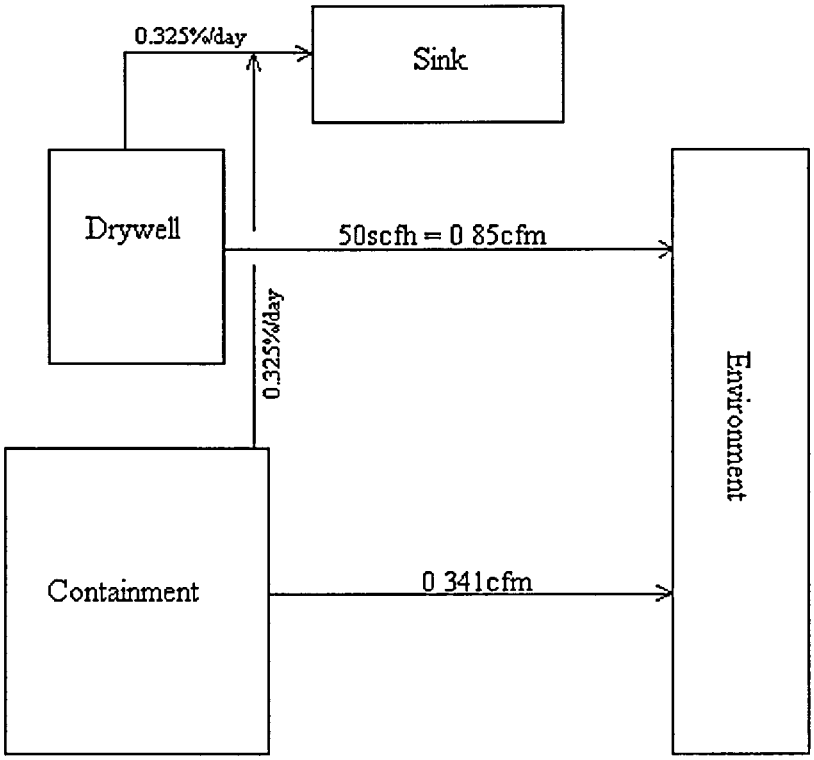


Figure 2c
Secondary Containment Bypass and MSIV Releases: 25 Minutes – 1.9 Hours

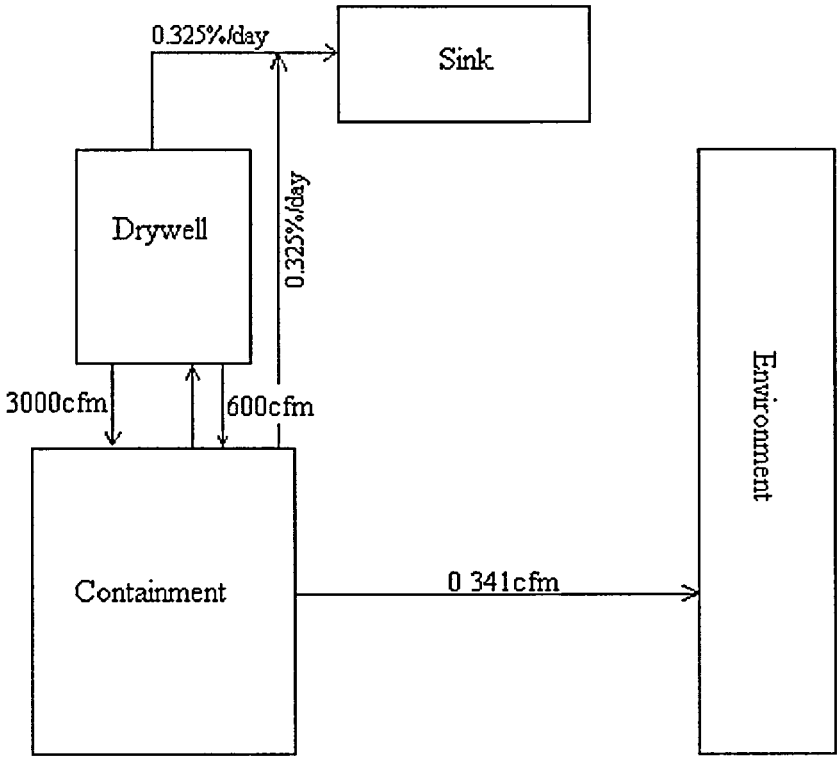


Figure 2d
Secondary Containment Bypass and MSIV Releases: 1.9 – 24 Hours

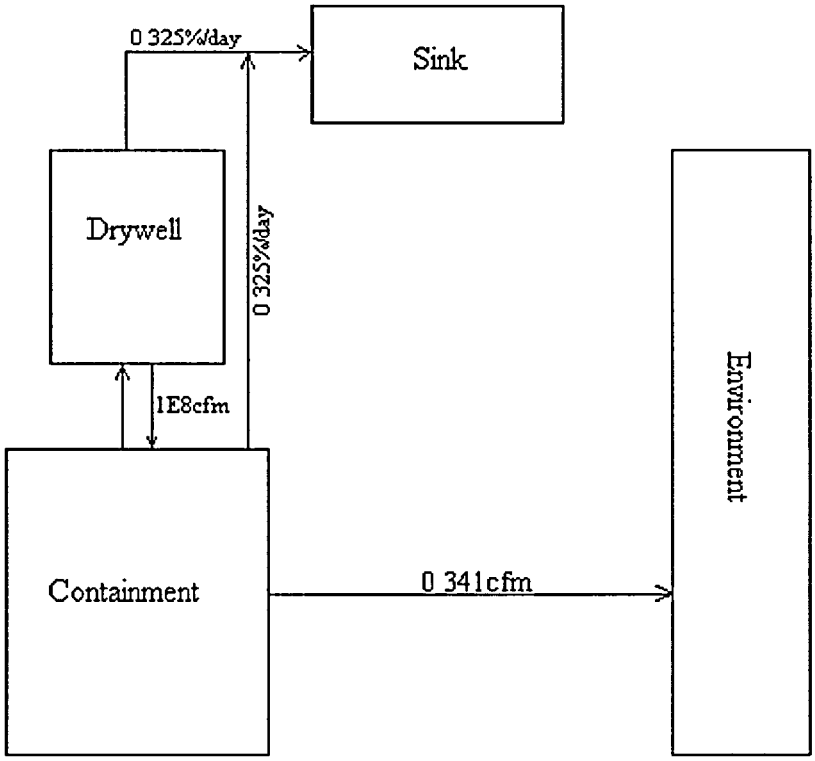


Figure 2e
Secondary Containment Bypass and MSIV Releases: 1 – 30 Days

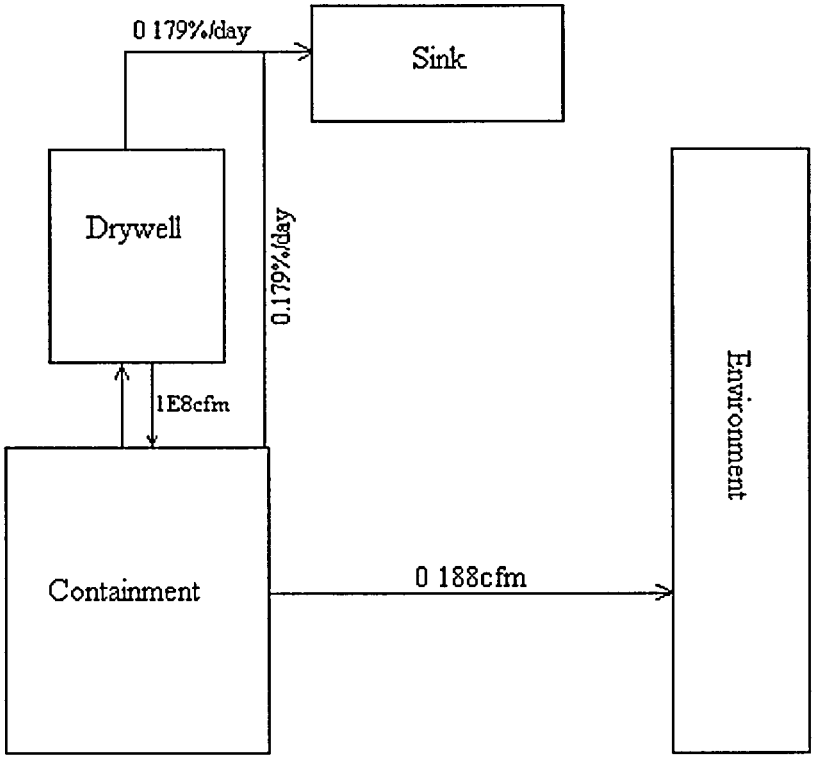


Figure 3a
Engineered Safety Features Liquid Leakage: 0 – 30 Minutes

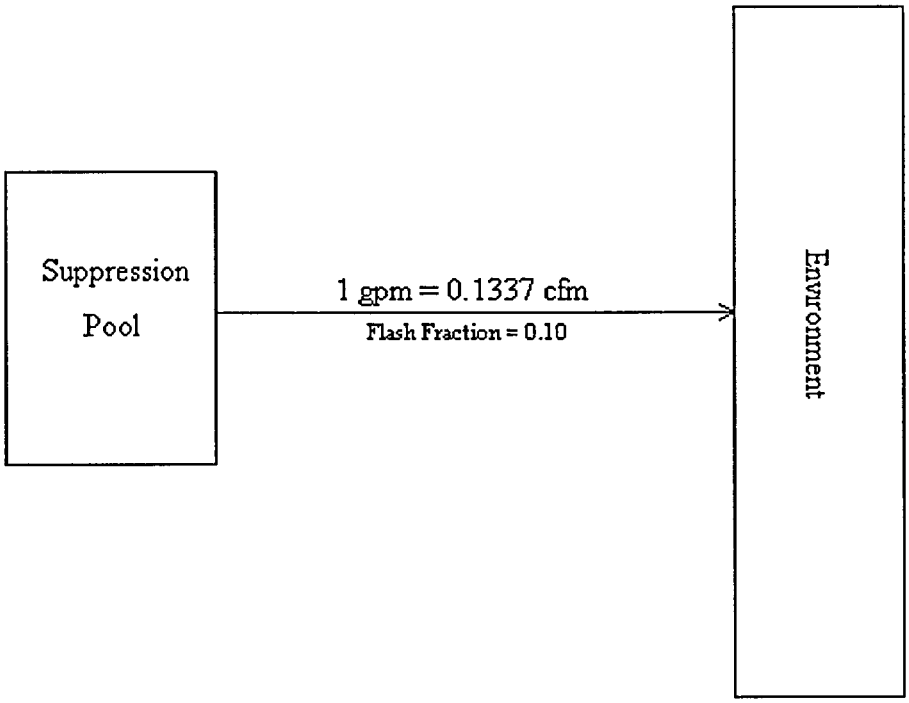


Figure 3b
Engineered Safety Features Liquid Leakage: 30 Minutes – 30 Days

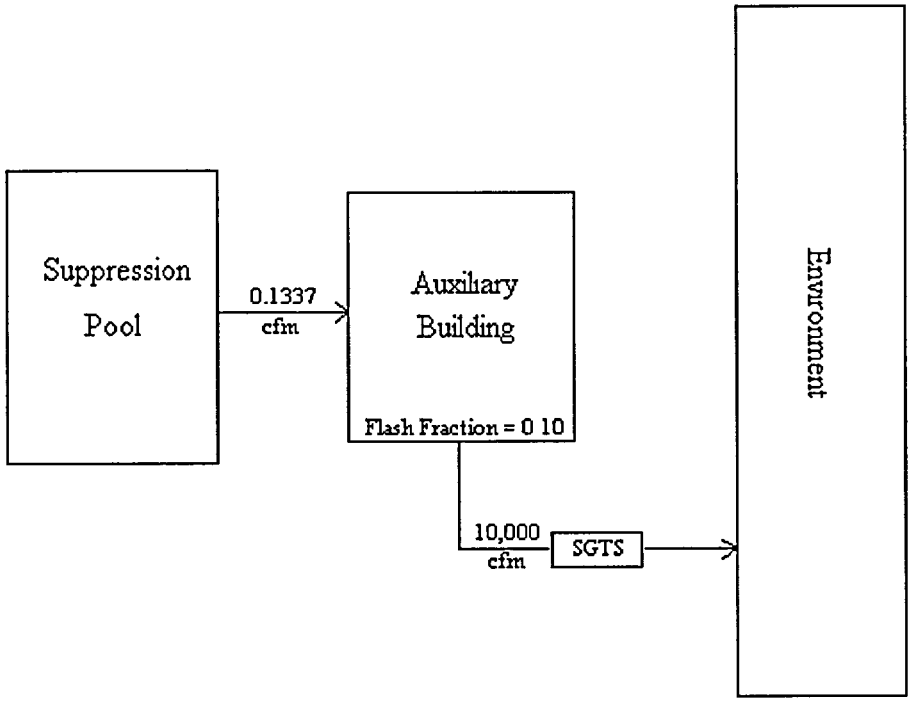


Figure 4a
Control Room Ventilation Model – Normal Mode

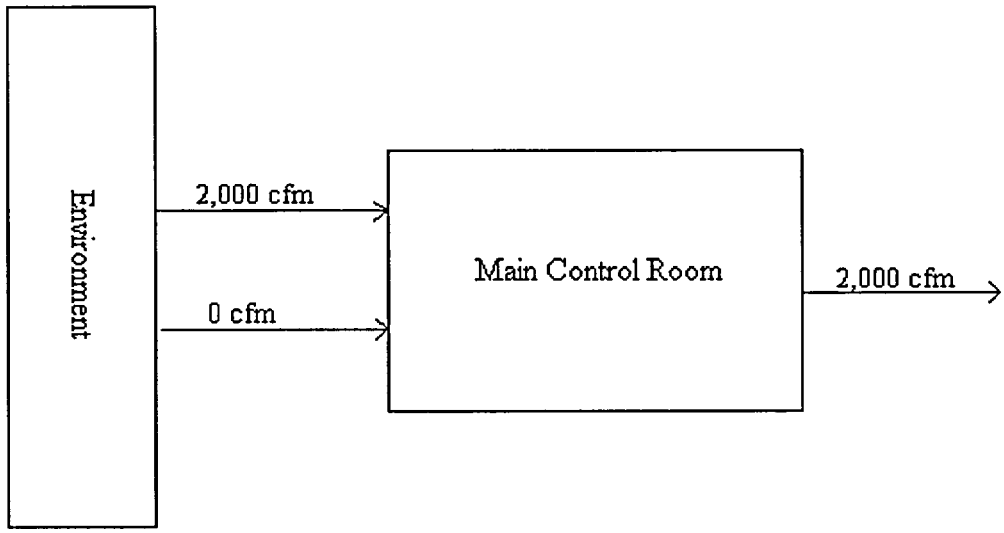
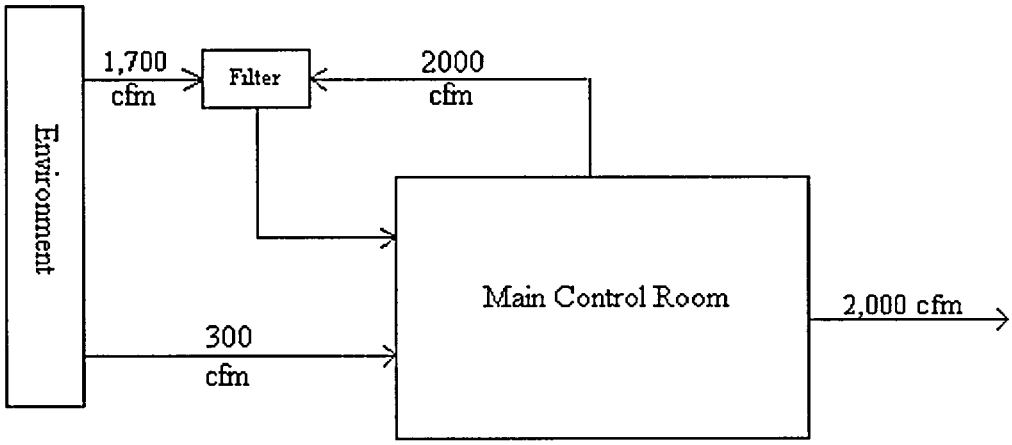


Figure 4b
Control Room Ventilation Model – Emergency Mode



Attachment 8

Letter RBG-46052

Fuel Handling Accident (FHA) and Light Load Drop Accident (LLA) Summary

Fuel Handling Accident (FHA) and Light Load Drop Accident (LLA) Summary

SCOPE

The Fuel Handling Accident (FHA) is postulated to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in a drop of a raised fuel assembly onto stored fuel bundles. Several FHA events have been evaluated including postulated scenarios for containment upper pool refueling operations as well as fuel building activities. The postulated FHA inside containment results in a larger number of fuel rods damaged as a result of a drop of a spent fuel bundle over the reactor core.

A bounding RBS FHA event involving the drop of a spent fuel assembly onto fuel assemblies in either the spent fuel pool or the reactor core has been analyzed. All fuel types used by RBS were evaluated (GE8, GE11, and Framatone Atrium 10 fuel). Based on conservative and limiting assumptions consistent with USNRC Regulatory Guide 1.183, the RBS bounding FHA event occurs in containment and results in a total of 122 GE 9x9 fuel rods damaged. This value was conservatively increased to 150 GE 9x9 rods. The failed rods gap activity is immediately released to the fuel building or containment atmosphere. Over a period of two hours, this accident activity is released into the atmosphere without crediting building mixing or dilution. No credit is taken for operation of either building's charcoal filtration units. Also, no credit is taken for filtration by the Main Control Room ESF charcoal filters. However, credit is taken for control room operations manual selection of the more favorable air intake in accordance with SRP 6.4 guidelines (i.e., the more favorable χ/Q is divided by four since RBS has dual air intake with manual selection).

The postulated accident activity is released as a "ground level release" from the fuel building or containment ventilation vent and dispersed to offsite and control room receptors according to plant specific atmospheric dispersion factors demonstrated as bounding for both release points. These plant specific offsite and control room dispersion factors have been shown to be conservative in comparison to those estimated per the criteria in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) respectively, as demonstrated in Table 1.

The Light Load Drop Accident (LLA) is closely related to the FHA. "Light loads" for RBS are $\leq$ 1200 lbs. This scenario involves dropping a load, other than fuel, onto irradiated fuel in either the containment or fuel buildings. Typically LLA scenarios are bounded by FHA results, however, RBS performed a conservative analysis to demonstrate that a drop of a light load could not cause dose limits to be exceeded. The bounding scenario was the drop of a light load in primary containment from the polar crane onto the reactor core. This drop results in a maximum of 247 GE 9x9 fuel rods damaged, however, the dose analysis conservatively used 300 rods to ensure that all fuel types are bounded. All other assumptions are consistent with the FHA analysis.

The radiological consequences of the event are evaluated against the acceptance criteria of 10CFR50.67 and GDC 19 of 10CFR50 Appendix A.

INPUTS AND ASSUMPTIONS

The inputs and assumptions used in the radiological consequence analysis of the FHA are listed in Tables 1 and 2. The assumptions used in the LLA dose analysis are presented in Table 4. These inputs are consistent with the requirements of RG 1.183.

RESULTS

The radiological consequences for the postulated RBS FHA event are summarized in Table 3.

The FHA event results in offsite and control room doses within the regulatory limits of 10CFR50.67 and GDC 19 of 10CFR50 Appendix A. No exceptions to Regulatory Guide 1.183 guidance were taken.

TABLE 1

COMPARISONS OF ATMOSPHERIC DISPERSION FACTORS

Location	Values Based on M&C/ RG 1.145 Methodology	ARCON96/PAVAN Fuel Building Ventilation/Truck Bay Doors (Notes 1,4)	ARCON96/PAVAN Containment Equipment Hatch (Note 1)
EAB			
• 0 – 2 hours	8.58E-04 sec./m^3	Note 2	6.52E-05 sec./m^3
LPZ			
• 0 – 8 hours	1.13E-04 sec./m^3	Note 2	6.62E-05 sec./m^3
• 8 – 24 hours	7.89E-05 sec./m^3		4.51E-05 sec./m^3
• 1 – 4 days	3.65E-05 sec./m^3		1.96E-05 sec./m^3
• 4 – 30 days	1.21E-05 sec./m^3		5.95E-06 sec./m^3
Main Control Room Normal Intake			
• 0 – 2 hours	1.62E-03 sec./m^3	1.09E-03 sec./m^3	1.21E-03 sec./m^3
• 2 – 8 hours	1.62E-03 sec./m^3	6.83E-04 sec./m^3	7.46E-04 sec./m^3
• 8 – 24 hours	1.20E-03 sec./m^3	3.17E-04 sec./m^3	3.39E-04 sec./m^3
• 1 – 4 days	4.05E-04 sec./m^3	2.34E-04 sec./m^3	2.65E-04 sec./m^3
• 4 – 30 days (Note 3)	6.48E-05 sec./m^3	1.95E-04 sec./m^3	2.20E-04 sec./m^3
Main Control Room Remote Intake			
• 0 – 2 hours	1.90E-03 sec./m^3	8.56E-04 sec./m^3	3.44E-04 sec./m^3
• 2 – 8 hours	1.90E-03 sec./m^3	5.53E-04 sec./m^3	2.27E-04 sec./m^3
• 8 – 24 hours	1.46E-03 sec./m^3	2.41E-04 sec./m^3	9.62E-05 sec./m^3
• 1 – 4 days	6.08E-04 sec./m^3	1.89E-04 sec./m^3	7.75E-05 sec./m^3
• 4 – 30 days	1.52E-04 sec./m^3	1.42E-04 sec./m^3	5.78E-05 sec./m^3

Note 1: The 5% χ/Q values are presented for comparison.

Note 2: Fuel Building Cask Handling Door values were not calculated for off-site locations.

Note 3: The Murphy & Campe values bound those recalculated using ARCON96 with the exception of the 4-30 day values. Since the FHA/LLA analyses assume a 2 hour release use of the higher values during this time frame would not impact the calculated results.

Note 4: The fuel building ventilation system exhaust vent was assumed for the RBS Control Room's Main Air Intake, and the truck bay doors was assumed for the remote air intake. Each release point was the most conservative location for the respective air intake.

TABLE 2
RBS FHA RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
I. Data and assumptions used to estimate radioactive source from postulated accident.	
1. Power Level	3100 MWt
2. Number of damaged rods (GE11) * All fuel types used by RBS were evaluated. GE11 (9x9 array) was determined to be bounding for RBS.	150
3. Total number of rods in core	46176 (Limiting GE 9x9 Fuel)
4. Core Activity available for release	Table 3
5. Radial peaking factor	2.00
6. Gap Activity Release Fractions* * RBS has verified that the fuel does not exceed the burnup requirements to meet the release fractions of Regulatory Guide 1.183.	Per Table 3 of RG 1.183
7. Release fission product species and chemical form	Per RG 1.183, Appendix B
8. Decay time	24 hours
II. Data and assumptions used to estimate activity released to the environment.	
1. 1. Building Release Rate	2 hour linear release rate
2. Halogen Decontamination Factor	200
III. Dispersion Data	
1. EAB X/Q Data 0-2 hrs	8 58E-04 sec/m^3
2. LPZ X/Q Data 0-8 hrs 8-24 hrs 1-4 days 4-30 days	1.13E-04 sec/m^3 7.89E-05 sec/m^3 3 65E-05 sec/m^3 1.21E-05 sec/m^3
3. Control Room X/Q Data* 0-20 min 20 min-8 hrs 8-24 hrs 1-4 days 4-30 days *RBS has dual Control Room air intakes and redundant radiation detectors within each intake. This dispersion data credits operator action at 20 minutes to select most favorable air intake.	1.62E-03 sec/m^3 4.05E-04 sec/m^3 3.00E-04 sec/m^3 1.01E-04 sec/m^3 1.62E-05 sec/m^3
IV. Control Room Parameters	
1. Free Air Volume	188,000 ft^3
2. Unfiltered In-leakage Rate	300 cfm
3. Outside Air Ventilation Rate	1700 cfm
4. CR ESF Iodine Filter Efficiency	0% (Not credited)
5. Control Room Breathing Rates and Occupancy Factors	Per RG 1.183

TABLE 3

RBS FHA CORE ACTIVITY
AT REACTOR SHUTDOWN (i.e., Decay Time = 0 hours)

Isotope	EOC Core Inventory (Ci/MWt)
I-131	2.631E+04
I-132	3.845E+04
I-133	5.502E+04
I-134	6.056E+04
I-135	5.195E+04
Kr-85	3.015E+02
Kr-85m	6.734E+03
Kr-87	1.292E+04
Kr-88	1.830E+04
Xe-133	5.528E+04
Xe-135	7.148E+03

TABLE 4

LIGHT LOAD DROP PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
Maximum Load Height – Polar Crane	El. 214.3'
Top of Fuel Bundles in Reactor Core	El. 133.5'
Maximum Drop Distance	80.8'
Load Weight	1200 lbs
Cladding Yield Strength	200 ft-lbs/GE 9x9 rod
Clad / Non-Fuel Mass Ratio	0.510
Kinetic Energy Absorbed by Fuel	100%
Total Kinetic Energy Available	96,600 ft-lb
Kinetic Energy Absorbed by Cladding	49,500 ft-lb
Number of Rods Damaged	247 GE 9x9 rods

TABLE 5

FHA AND LLA RADIOLOGICAL CONSEQUENCES

Receptor	Regulatory Limit (REM TEDE)	FHA Dose (REM TEDE)	LLA Dose (REM TEDE)
EAB	6.3	2.5	5.0
LPZ	6.3	0.4	0.7
Control Room	5	1.7	3.3

TABLE 6

FHA ACTIVITY RELEASED TO ENVIRONMENT

Isotope	Gap Activity(1) [Ci] (t=0 hrs)	Gap Activity(2) [Ci] (t=24 hrs)	Activity Released to Environment(3) [Ci]
I-131	4.20E+04	3.85E+04	1.90E+02
I-132	3.90E+04	2.82E+01	1.00E-01
I-133	5 50E+04	2.47E+04	1.20E+02
I-134	6.10E+04	3.50E-04	8.50E-07
I-135	5.20E+04	4.20E+03	1.90E+01
I-136	N/A	N/A	N/A
Kr-83m	N/A	N/A	N/A
Kr-85	6.10E+02	6.10E+02	6.10E+02
Kr-85m	6.80E+03	1.66E+02	1.40E+02
Kr-87	1.30E+04	2.71E-02	1.60E-02
Kr-88	1.80E+04	5.14E+01	4.10E+01
Kr-89	N/A	N/A	N/A
Xe-131m	N/A	N/A	N/A
Xe-133m	N/A	N/A	N/A
Xe-133	5.60E+04	4.91E+04	4.90E+04
Xe-135m	N/A	N/A	N/A
Xe-135	7.20E+03	1.16E+03	1.10E+03
Xe-137	N/A	N/A	N/A
Xe-138	N/A	N/A	N/A

Note 1: The gap activity column for t=0 represents the values actually utilized in the RADTRAD files.

Note 2: The gap activities for 24 hours were estimated by decaying the values found in column 1. The actual analysis utilized RATRAD to decay the isotopes.

Note 3: The activity released to the environment is taken directly from the RADTRAD output file.

Attachment 9

Letter RBG-46052

Control Rod Drop Accident (CRDA) Summary

CONTROL ROD DROP ACCIDENT (CRDA) SUMMARY

SCOPE

The plant design basis Control Rod Drop Accident (CRDA) is a postulated event in which a high worth control rod drops from its fully inserted or intermediate position in the core. The removal of large negative reactivity from the core results in a localized power excursion. For the CRDA accident scenario GE8 fuel was found to be bounding. Based on conservative and limiting assumptions consistent with USNRC Regulatory Guide 1.183, the RBS postulated CRDA at full power results in a total of 850 GE8 fuel rods damaged.

The postulated accident activity is released at ground level from the plant condenser and dispersed to offsite and control room receptors according to plant specific atmospheric dispersion factors. These plant specific offsite and control room dispersion factors were determined in accordance with Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) guidance, respectively.

Control room doses are estimated for two CRDA events:

1. A design basis case based on Regulatory Guide 1.183 assumptions. Source term is conservatively based on 100% power operations with activity released via the plant condenser. No credit is taken for the Control Room ESF charcoal filters. Activity is released instantaneously from the reactor fuel to the coolant.

2. Low power operation. Limited CRDA that does not result in Main Steam Line Radiation Monitor (MSLRM) Trip. Activity release via the condenser Mechanical Vacuum Pumps at 4000 cfm until isolated per operator action at 20 minutes. Manual initiation of the Control Room ESF charcoal filters is credited 20 minutes into the event. A 10 second "burst" release is assumed from the reactor fuel in accordance with NEDO-31400A, *Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Radiation Monitor*, assumptions.

The radiological consequences of the event are evaluated against the acceptance criteria of 10CFR50.67.

INPUTS AND ASSUMPTIONS

The inputs and assumptions used in the radiological consequence analysis of the RBS CRDA are listed in Tables 1 and 2. These inputs are consistent with the requirements of RG 1.183.

In the low power operation CRDA event, activity release assumptions are established to maximize both the offsite and Control Room doses. Offsite doses for this case are estimated based cladding damage only. The dose rate to the MSLRM was calculated to be 8.4 R/hr which corresponds to a "design basis" N-16 concentration of 300 uCi/g (See USAR Table 11.1-1) which only occurs during Hydrogen Water Chemistry (HWC) operation (the maximum N-16 concentration Normal Water Chemistry, NWC, is 50 uCi/g). The fuel damage is based on a Main Steam Line Radiation Monitor setpoint of 30 R/hr which corresponds to 25 R/hr (3.0 x HWC dose rates) plus 20% for instrument uncertainty. The dose rate from 1 GE 8x8 rod (assuming a

"burst" release) was determined to be 0.6 R/hr. Background radiation is neglected this corresponds to 50 GE 8x8 rods being damaged. The MSLRM setpoints are based on 1.5x and 3.0x "full power background" values. The HWC system affects dose rates in the steam affected areas of the plant by as much as a factor of 6 (50 uCi/g vs. 300 uCi/g). The assumptions used justify basing the MSLRM isolation setpoint (HIGH-HIGH) on 6x design basis HWC dose rates which correspond to 18.0x NWC dose rates. Similarly, the HIGH setpoint will be based on 1.5x HWC design basis dose rates which corresponds to 9.0x NWC values.

RESULTS

The radiological consequences for the postulated RBS CRDA events are summarized in Table 3.

The Full Power and Low Power CRDA events result in offsite and control room doses within the regulatory limits of 10CFR50.67. No exceptions to Regulatory Guide 1.183 guidance were taken.

TABLE 1

RBS CRDA RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
I. Data and assumptions used to estimate radioactive source from postulated accident.	
1. Power Level	3100 MWt
2. Number of damaged rods 100% Power Event Low Power Event (gap release)	 850 GE 8x8 50 GE 8x8
3. Total rods in core GE 8x8 GE 9x9	 38,688 (62 rods per assembly) 46,176 (74 rods per assembly)
4. Number of assemblies damaged Design Basis – Maximum Fuel Damage (based on 8x8) Limited CRDA – No MVP trip (Based on 8x8) Note: For the CRDA scenario GE8 fuel is bounding. This is confirmed each reload cycle.	 850/62 = 13.7 50 / 62 = 0.8
5. Core Activity available for release	Table 2
6. Radial Peaking Factor	2.00
7. Assumed % fuel melt Design Basis – Maximum Fuel Damage Limited CRDA – No MVP trip	 100% 0%
8. Gap Activity Release Fractions	Per RG 1.183, Table 3 and Appendix C 10% noble gases, 10% iodines, 12% alkali metals
9. Fuel Melt Release Fractions	Per RG 1.183, Appendix C 100% noble gases, 50% Iodines
10. Fuel Release Duration Design Basis – Maximum Fuel Damage Limited CRDA – No MVP trip	 Instantaneous 10 sec. burst
II. Data and assumptions used to estimate activity released to the environment.	
1. Condenser Leak Rate Design Basis CRDA Limited CRDA – No MVP trip	 Per RG 1.183, Appendix C. 1% per day for 24 hours 4000 cfm for 20 minutes, 1% per day for next 24 hours
2. Condenser Iodine Release Fractions	Per RG 1.183, Appendix C 97% Elemental, 3% Organic
3. Condenser Radioactive Decay During Holdup	Credited
4. Condenser Volume	106,460 ft^3

Description of Input/Assumption	Design Basis Input and/or Assumption
III. Dispersion Data	
1. EAB X/Q Data	
0-2 hrs	7.51E-04 sec/m^3
2. LPZ X/Q Data	
0-8 hrs	7.79E-05 sec/m^3
8-24 hrs	5.23E-05 sec/m^3
1-4 days	2.21E-05 sec/m^3
4-30 days	6.40E-06 sec/m^3
3. Control Room X/Q Data	
Main Air Intake*	3.03E-03 sec/m^3
0-2 hrs	2.47E-03 sec/m^3
2-8 hrs	1.05E-03 sec/m^3
8-24 hrs	9.01E-04 sec/m^3
1-4 days	6 74E-04 sec/m^3
4-30 days	
Remote Air Intake	
0-2 hrs	3.69E-04 sec/m^3
2-8 hrs	3.08E-04 sec/m^3
8-24 hrs	1.30E-04 sec/m^3
1-4 days	1.07E-04 sec/m^3
4-30 days	7.15E-05 sec/m^3
Flow Biased Values**	
0-2 hrs	5.31E-04 sec/m^3
2-8 hrs	4.63E-04 sec/m^3
8-24 hrs	1.85E-04 sec/m^3
1-4 days	1.58E-04 sec/m^3
4-30 days	1.16E-04 sec/m^3
*The "design basis" case assumes the main air intake values for the duration of the event since no credit is taken for operators initiating the ESF filters. **RBS has dual Control Room air intakes and redundant radiation detectors within each intake. This dispersion data credits operator action and weights the values based on assumed flow rates (similar to the LOCA analysis).	
IV. Control Room Parameters	
1. Free Air Volume	188,000 ft^3
2. Unfiltered In-leakage Rate	300 cfm
3. Outside Air Ventilation Rate	1700 cfm

Description of Input/Assumption	Design Basis Input and/or Assumption
4. Intake Iodine Filter Efficiency	
Design Basis CRDA Aerosol Elemental and Organic Limited CRDA Aerosol Elemental and Organic	 0% 0% 99% 98%
5. Time for Control Room Ventilation Isolation per Operator Action. Design Basis CRDA Limited CRDA	 Not credited 20 minutes
6. Emergency Mode Recirculation Rate (Post-isolation Mode)	2000 cfm
7. Control Room Breathing Rates and Occupancy Factors	Per RG 1.183

TABLE 2

RBS CRDA CORE ACTIVITY

Isotope	EOC Core Inventory (Ci/MWt)
I-131	2.631E+04
I-132	3.845E+04
I-133	5.502E+04
I-134	6.056E+04
I-135	5.195E+04
Kr-85	3.015E+02
Kr-85m	6.734E+03
Kr-87	1.292E+04
Kr-88	1.830E+04
Xe-133	5.528E+04
Xe-135	7.148E+03
Cs-134	5.357E+03
Cs-136	1.179E+03
Cs-137	3.324E+03
Rb-86	4.696E+01

TABLE 3

CONTROL ROD DROP ACCIDENT RADIOLOGICAL CONSEQUENCES

Receptor	Regulatory Limit (TEDE)	100% Power Event Dose (TEDE)	Low Power Event Dose (TEDE)
EAB	6.3	0.9	4.7
LPZ	6.3	0.4	0.5
Control Room	5	4.3	1.3

TABLE 4

CRDA ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	100% Power Event	Low Power Event
Kr-85	4.09E+02	1.29E+02
Kr-85m	9.14E+03	2.88E+03
Kr-87	1.31E+03	4.92E+03
Kr-88	4.17E+03	7.40E+03
Rb-86	7.51E-04	2.41E-03
I-131	1.71E+02	1.12E+02
I-132	3.56E+01	1.54E+02
I-133	2.55E+02	2.33E+02
I-134	2.09E+01	2.20E+02
I-135	1.27E+02	2.16E+02
Xe-133	7.02E+04	2.36E+04
Xe-135	4.48E+03	3.00E+03
Cs-134	8.72E-02	2.74E-01
Cs-136	1.87E-02	6.04E-02
Cs-137	5.42E-02	1.70E-01

Attachment 10

Letter RBG-46052

Main Steam Line Break (MSLB) Outside of Primary Containment Summary

MAIN STEAM LINE BREAK ACCIDENT (MSLB) OUTSIDE OF PRIMARY CONTAINMENT SUMMARY

SCOPE

The plant design basis Main Steam Line Break Accident (MSLB) assumes an instantaneous guillotine break at one of the four main steam lines outside containment at a location downstream of the outermost isolation valve. The break results in mass loss from both ends of the break. The flow from the upstream side of the break is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictor in the three unbroken lines. Subsequent closure of the Main Steam Isolation Valves (MSIVs) further limits the flow when the valve flow area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached. The mass release used is based on initial licensing of the plant and bounds power uprated conditions.

No credit was taken for the Control Room ESF charcoal filter trains No fuel damage is predicted and thus the postulated release halogen activity is based on the maximum coolant activity allowed by Technical Specifications as required by USNRC Regulatory Guide 1.183, Appendix D. The noble gas activity is based on an offgas release rate of 310,000 uCi/sec. (after 30 minutes decay) which conservatively bounds the 290 mCi/sec. allowed per Technical Specifications. The alkali metals design basis coolant concentrations were also increased by 2% to account for uncertainties in the core thermal power level. The analysis conservatively assumed that 100% of the halogen and alkali metal activity present in the released coolant was transported to the environment.

The postulated accident activity is released as a "ground level release" from the Main Steam Tunnel (MST) blowout panel and dispersed to offsite and control room receptors according to plant specific atmospheric dispersion factors. These plant specific offsite and control room dispersion factors were determined in accordance with Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) respectively.

Consistent with RG 1.183, Appendix D, Offsite and Control room doses are estimated for two MSLB events:

1. Primary coolant iodine concentration corresponding to an assumed maximum pre-accident spike of 4 μCi/gm dose equivalent I-131 per plant Technical Specifications.

2. Primary coolant iodine concentration corresponding to the Technical Specifications' maximum equilibrium value of 0.2 μCi/gm dose equivalent I-131.

The radiological consequences of the event are evaluated against the acceptance criteria of 10CFR50.67.

INPUTS AND ASSUMPTIONS

The inputs and assumptions used in the radiological consequence analysis of the RBS MSLB are listed in Tables 1 and 2. These inputs are consistent with the requirements of RG 1.183.

RESULTS

The radiological consequences for the postulated RBS MSLB events are summarized in Table 3.

The MSLB events result in offsite and control room doses within the regulatory limits of 10CFR50.67. No exceptions to Regulatory Guide 1.183 guidance were taken.

TABLE 1

RBS MSLB RADIOLOGICAL CONSEQUENCE ANALYSIS PARAMETERS

Description of Input/Assumption	Design Basis Input and/or Assumption
I. Data and assumptions used to estimate radioactive source from postulated accident.	
1. Power Level	3100 MWt
2. Maximum Pre-accident Spike Iodine Concentration	4 μCi/gm DE I-131
3. Maximum Equilibrium Iodine Concentration	0.2 μCi/gm DE I-131
4. Noble Gas Source Term	Based on 310,000 μCi/sec at 30 minutes, corrected to time equal zero.
5. Alkali Metals	Reactor coolant activity design concentration ratioed to account for 102% power.
II. Data and assumptions used to estimate activity released to the environment.	
1. Mass Release (Note this data corresponds to that calculated for initial licensing of the plant. Analyses demonstrate these values bound the hot standby conditions for Power Uprated conditions.)	Steam, 11620 lbm Liquid, 68942 lbm
2. Iodine Carryover Fraction	4%
3. Break Isolation Time	5.5 seconds
4. Building Release Rate	Instantaneous ground level release with no credit for plateout, holdup, or dilution.
5. Iodine Species Release Fractions to Environment	Per RG 1.183, Appendix D, 95% Aerosol, 4.85 % Elemental, 0.15% Organic
6. Activity Released to Environment	Table 2
III. Dispersion Data	
1. EAB X/Q Data 0-2 hrs	 8.25E-04 sec/m^3
2. LPZ X/Q Data 0-8 hrs 8-24 hrs 1-4 days 4-30 days	 1.11E-04 sec/m^3 7.79E-05 sec/m^3 3.60E-05 sec/m^3 1.19E-05 sec/m^3
3. Control Room X/Q Data* 0-8 hrs 8-24 hrs 1-4 days 4-30 days *Since no credit is taken for the ESF filter trains the X/Q values assumed in this analysis were based on the Main Air Intake.	 3.64E-03 sec/m^3 2.69E-03 sec/m^3 1.46E-03 sec/m^3 2.73E-04 sec/m^3
IV. Control Room Parameters	
1. Free Air Volume	188,000 ft^3
2. Unfiltered In-leakage Rate	300 cfm

Description of Input/Assumption	Design Basis Input and/or Assumption
3. Outside Air Ventilation Rate	1700 cfm
4. Emergency Mode Filtered Intake/Unfiltered Inleakage Rate (1700 cfm ventilation rate + 300 inleakage rate)	2000 cfm
5. Control Room Breathing Rates and Occupancy Factors	Per RG 1.183

TABLE 2

RBS MSLB ACTIVITY RELEASED TO ENVIRONMENT

Isotope	0.2μCi/gm DE I-131 Case (Ci)	4μCi/gm DE I-131 Case (Ci)
I-131	1.70E+00	3.40E+01
I-132	2.51E+01	5.02E+02
I-133	2.27E+01	4.53E+02
I-134	3.97E+01	7.94E+02
I-135	2.19E+01	4.37E+02
Cs-134	5.42E-03	5.42E-03
Cs-136	3.51E-03	3.51E-03
Cs-137	1.40E-02	1.40E-02
Kr-85m	5.22E-02	5.22E-02
Kr-85	1.63E-04	1.63E-04
Kr-87	1.79E-01	1.79E-01
Kr-88	1.79E-01	1.79E-01
Xe-133	6.58E-02	6.58E-02
Xe-135	1.95E-01	1.95E-01

TABLE 3

MAIN STEAM LINE BREAK ACCIDENT RADIOLOGICAL CONSEQUENCES

Case	EAB Dose (REM TEDE)	LPZ Dose (REM TEDE)	Regulatory Limit (REM TEDE)
4μCi/gm DE I-131	1.4	0.2	25
0.2μCi/gm DE I-131	<0.1	<0.1	2.5

Case	Control Room Dose (REM TEDE)	Regulatory Limit (REM TEDE)
4μCi/gm IDE	2.2	5
0.2μCi/gm IDE	0.2	5